BEAVER VALLEY POWER STATION UNIT 2

STARTUP REPORT SUPPLEMENT 2

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

TABLE OF CONTENTS

LIST OF ABBREVIATIONS

ANSI/ANS	American National Standards Institute/American Nuclear Society
B&W	Babcock & Wilcox
BVPS	Beaver Valley Power Station
db	Decibels
dba	Decibels Absolute
F	Degrees Fahrenheit
FSAR	Final Safety Analysis Report
I&C	Instrumentation and Control
LPMS	Loose Parts Monitoring System
MPH	Miles Per Hour
RCS	Reactor Coolant System
RTD	Resistance Temperature Detector
Th	Reactor Coolant Hot Leg Temperature

1.0 INTRODUCTION

The second supplement to the Beaver Valley Power Station Unit 2 (BVPS-2) Startup Report covers the period of testing deferred after Commercial Operations between February 15, 1988 and May 15, 1988. This supplement report is prepared in accordance with Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A, Technical Specifications", and addresses the results of deferred testing identified in the BVPS-2 Final Safety Analysis Report during this period. Those tests which were deferred, but not completed by May 15, 1988, will also be identified in this report, but a comprehensive discussion on these tests will be provided when they are completed. Another supplement report will cover those tests completed subsequent to May 15, 1988.

The Unit returned to approximately 100% power on February 14, 1988 following a two week maintenance outage in Mode 5. On February 17, 1988, the Unit reduced power to approximately 75% power to allow the "D" condenser waterbox to be removed from service for leak inspection. While waterbox leak inspections continued, the Unit reduced power to approximately 60% for maintenance on the "B" Feedwater Pump. No condenser waterbox leaks were discovered. The Unit returned to 100% power on February 22, 1988 following completion of feedwater pump maintenance. The Unit ran at approximately 100% power for 42 consecutive days. On April 4, 1988 at 0847 hours, the reactor tripped due to a low RCS flow in the "A" loop. The Unit returned to approximately 100% power on A[illegible], 1988, and continued full power operation through the remainder of this report's period. Section 3.0 of this supplement report contains a Post-Commercial Operations Chronolog for additional information.

2.0 DEFERRED POST-COMMERCIAL OPERATIONS TEST PROCEDURES

The test procedures found in this section could not be completed prior to Commercial Operations, and were deferred as concurred with by the Onsite Safety Committee and formally identified to the NRC. All remaining test procedures which could not be completed in the time frame of this supplement report have a scheduled completion date of the first refueling. This section will discuss test results for those test procedures completed after Commercial Operations between February 15, 1988 and May 15, 1988. Although SOV-2.45A.01, "Loose Parts Monitoring System Test", was completed prior to Commercial Operations, evaluation of data by the Loose Parts Monitoring System vendor resulted in troubleshooting after Commercial Operations which will also be discussed in this section. Another supplement report will be issued covering the period of May 15, 1988 to August 13, 1988.

2.1 Verification of Plant Performance Following Plant Load Rejection/Trip From Power (IST-2.04.06)

Beaver Valley Power Station, Unit 2, was designed with the capability of sustaining a 100% full load rejection without a turbine trip or a reactor trip. This test performed a full load rejection from approximately 100% power prior to Commercial Operations, but did not satisfy all acceptance criteria. A detailed discussion about the full load rejection can be found in Section 7.2.5 of the original Startup Report.

The other portion of this test was to perform a turbine trip from 100% power. Data obtained from an inadvertent turbine trip from approximately 100% power prior to Commercial Operations was evaluated and found to be acceptable to satisfy the performance of the turbine trip portion of this test. A discussion about the turbine trip can be found in Section 7.2.5 of the original Startup Report.

As discussed in Section 7.2.5 of the original Startup Report, data collected during the inadvertent turbine trip and during the unsuccessful full load rejection turbine trips, could not be used for the Reactor Coolant Hot Leg Resistance Temperature Detector (RTD) time response evaluation. Retest 04-002 was generated to obtain the required RTD response time during the next inadvertent reactor trip from 100% power. Since the plant was declared Commercial on the day this retest was initiated, the retest and IST-2.04.06 were both deferred until prior to the first refueling. I&C was directed to connect a memory type strip chart recorder to the Th RTD outputs at the primary process racks to gather the required data for Westinghouse during the next reactor trip from 100% power. The recorder would automatically start upon a reactor trip, and Westinghouse would use this information to determine plant RTD time response. An inadvertent reactor trip occurred on November 17, 1987, subsequent to Commercial Operations, but prior to installing the strip chart recorder. Consequently, the RTD response time data was not obtained during this trip.

The Startup Report, Supplement 1, Section 2.3 discusses the results of data obtained during an inadvertent reactor trip from approximately 96% power on January 27, 1988. This data was evaluated by Westinghouse and determined not usable for the purpose of RTD response time evaluation. On April 4, 1988, an inadvertent reactor trip from approximately 100% power occurred due to a low RCS flow in the "A" loop. Since the analysis assumptions for the January 27, 1988 trip were the same for this trip, the data was determined not usable for the purpose of RTD response time evaluation. The analysis assumptions for development of RTD response time criteria is that the full reactor coolant pump flows are maintained in all loops for the duration of the required RTD response time (7-10 seconds). The flow increase or decrease in any one loop can thus affect the RTD bypass flow and can invalidate the acceptance criteria.

At the time of this report, the data required by Retest 04-002 had not been obtained. Until this data can be obtained during another inadvertent reactor trip from approximately 100% power, the plant will continue to operate with reduced overpower and overtemperature setpoints. The results of Retest 04-002 will be discussed in another supplement report subsequent to completion of the retest.

2.2 Verification of Reactor Plant Setpoints (PO-2.01A.03)

The initial reactor plant setpoints were verified by using the latest plant documentation and/or actual instrumentation module settings ensuring the equipment was aligned for the initial startup setpoint values. All setpoints fell within the required Technical Specification limitations. Final setpoint verification was started with the plant at 100% power, but could not be completed prior to Commercial Operations because several systems did not have the final setpoint values in place. The remainder of this test was deferred with a required completion date of the first refueling.

At the time of this supplement report, gathering of final setpoint values was still in progress. The results of final setpoint verification will be discussed in another supplement report after obtaining all final setpoint values.

2.3 Fuel Pool Cooling System Test (PO-2.20.01)

This test was satisfactorily completed prior to core load with the exception of Retest 20-003 which will measure Fuel Pool Cooling Pump 21A vibration the next time the Spent Fuel Pool is filled. Engineering resolution for high pump vibration during original testing modified the motor support stand by welding on stiffening plates. The pump motor was retested yielding acceptable vibration levels. The pump and motor were then recoupled, but the Spent Fuel Pool had already been drained. The Spent Fuel Pool will not be refilled until prior to the first refueling so the test was deferred with a required completion date of prior to the first refueling.

At the time of this supplement report, refilling of the Spent Fuel Pool had not yet been started. The results of Retest 20-003 will be discussed in another supplement report subsequent to completion of the retest.

2.4 Spent Fuel Pool and Refueling Cavity Leak Test (PO-2.20.02)

This test was satisfactorily completed prior to core load with the exception of Retests 20-001 and 20-002. Retest 20-001 was generated when the gates between the Fuel Pool and Cask Area and between the Fuel Pool and Fuel Transfer Canal exhibited considerable leakage at the bottom seals. Repairs were made to both gates but the retest could not be performed until the next time the Spent Fuel Pool was filled. Retest 20-002 was generated when the Spent Fuel Pool instrumentation test could not be performed due to unavailability of the Spent Fuel Pool level transmitters. This retest required the Spent Fuel Pool to be filled to its high level. The Spent Fuel Pool will not be refilled until prior to the first refueling so both retests were deferred with a required completion date of prior to the first refueling.

At the time of this supplement report, refilling of the Spent Fuel Pool had not yet been started. The results of Retests 20-001 and 20-002 will be discussed in another supplement report subsequent to completion of the retests.

2.5 Fuel Handling Equipment Test (PO-2.66.01)

This test verified control logic and demonstrated operability of the Fuel Handling Equipment and provided a functional demonstration of a simulated fuel transfer between the Fuel Building and Reactor Containment. Fuel Handling Equipment testing was satisfactorily completed to support core load although portions of the test were not done due to unavailability of equipment. Those portions of the test not available for testing prior to core load were completed prior to Commercial Operations except for testing of the RCC Change Fixture (Retest 06-005). A discussion on that portion of the test completed prior to Commercial Operations can be found in Section 8.4 of the original Startup Report. Testing of the RCC Change Fixture (Retest 06-005) was deferred with a required completion date of prior to the first refueling.

At the time of this supplement report, testing of the RCC Change Fixture had not yet been started. The results of Retest 06-005 will be discussed in another supplement report subsequent to completion of the retest.

2.6 Cranes and Lifting Equipment Test (PO-2.66.02)

Control logic testing for the Reactor Containment Polar Crane and operability of its auxiliary hoist under load was successfully completed prior to core load. Polar Crane main hoist operability under load was checked during its load test and satisfactorily completed during reactor vessel head installation. Retest 66-007 was generated to test an indicating light on the telescoping work platform which was inoperative when in the maximum down position. Outstanding rework on the Spent Fuel Cask Crane was not completed prior to core load because it was thought to be safer to do it without fuel in the Fuel Storage area. Retest 66-007 and the remainder of the test which tests the Spent Fuel Cask Crane, were deferred with a required completion date of prior to the first refueling.

At the time of this supplement report, electrical maintenance repair of the indicating light on the Polar Crane telescoping work platform, and testing of the Spent Fuel Cask Crane had not yet been performed. The results of Retest 06-007 and Spent Fuel Cask Crane testing will be discussed in another supplement report subsequent to completion of testing.

2.7 Verification of Performance Calculations (SOV-2.05A.03)

This test was performed at 30, 50, 75 and 100% power plateaus during Startup, and verified that the computer-generated performance calculations, Nuclear Steam Supply System (NSSS), and Balance of Plant (BOP) programs were accurate at these power levels. Deficiencies at lower power levels were cleared at subsequent power levels or Maintenance Work Requests were issued against deficiencies to perform troubleshooting with any required retests to be performed by existing plant procedures.

This procedure only satisfied a portion of the BVPS-2 Final Safety Analysis Report (FSAR) test objective that was required to be performed. This limited scope was justified by the fact that a comprehensive set of initial operating procedures for verifying the accuracy of the plant performance calculations was being developed and performed by the Computer Integration Group. These tests perform thorough verifications of all the programs and calculations related to the plant performance applications of the process computer system. The tests are designed in accordance with accepted software validation and verification standards, using test data sets to prove the accuracy of the program output. The accuracy of the field sensors was verified by the Phase 1 testing program. The accuracy of the raw value conversion into engineering units was verified by SOV-2.05A.03 as mentioned above. The total of final test results after completion, will fully satisfy all FSAR test objectives. The Computer Integration Group is continuing the testing per the Initial Operating Procedures. Deferral to beyond Commercial Operations was approved with the estimated completion date for this testing prior to initial criticality after the first refueling.

At the time of this supplement report, testing by the Computer Integration Group had not yet been completed. All test results will be discussed in another supplement report following completion of all testing.

2.8 Solid Waste Disposal System Test (SOV-2.18.01)

This test was partially performed prior to core load and deferred until needed to provide solid waste disposal capabilities at BVPS-2. As of Commercial Operations, only a portion of the test was satisfactorily completed. A discussion on that portion completed prior to Commercial Operations can be found in Section 8.8 of the original Startup Report. Section 2.13 of Supplement 1 to the Startup Report discusses testing completed on the Solid Waste Disposal System after Commercial Operations and prior to February 15, 1988.

Verification that Condensate Demineralizer Clamshell Filter Sludge could be transferred from the Clamshell Filter Sludge Tank to the Decant Tank, that free standing water could be removed from the contents of the Decant Tank, and that Sludge could be transferred from the Decant Tank to the Drumming Station in precise volumes was successfully completed on March 28, 1988.

Testing of the Solid Waste Disposal Drumming Station is all that remains to be completed and was still in progress during the time period covered by this supplement report. The remainder of the test will be completed as needed to provide solid waste disposal capabilities at BVPS-2. At present, the ability to dispose of radioactive solid waste is still possible through cross-ties with BVPS-1. The results will be discussed in another supplement report upon completion of remaining testing.

2.9 Automatic Steam Generator Water Level Control Test (SOV-2.24C.01)

This test was performed at various modes and power levels during the Startup Testing Program. A detailed discussion on that portion of the test completed prior to Commercial Operations can be found in Section 7.3.3 of the original Startup Report.

During Startup at the 30, 50, 75, and 100% power plateaus, each steam and feedwater flow transmitter output signal was compared to the associated calorimetric flow computed from test gauges in accordance with IST-2.02.06, "Thermal Power Calorimetric Test". At each power level, the feedwater and steam flow rates which were calculated did not agree with the calorimetric flow rate criteria. A copy of the test data was forwarded to I&C to make the necessary adjustments to both the feedwater and steam flow transmitters. These adjustments and the remainder of the test were not completed prior to Commercial Operations, and were deferred with a required completion date of the first refueling. The last section in this test will obtain additional flow data at 100% power to determine the need for further adjustments, and to demonstrate repeatability of the feedwater and steam flow transmitters at 30, 50, 75, and 100% power during a subsequent plant startup.

At the time of this report, I&C had not made the necessary adjustments to both the feedwater and steam flow transmitters. This still precludes the completion of the remainder of this test. Note that I&C had determined that there was no problem with a steam flow/feed flow mismatch and recommended that plant operation continue until final adjustments and checks can be performed. Test results will be discussed in another supplement report upon completion of the remainder of testing.

2.10 Steam Generator Blowdown System Test (SOV-2.25.01)

This test was completed prior to Commercial Operations except for clearing of minor deficiencies which occurred prior to Commercial Operations, and verification of operating parameters for the Cleanup Ion Exchangers and the "B" Steam Generator Blowdown Evaporator. A discussion on that portion of the test completed prior to Commercial Operations can be found in Section 8.11 of the original Startup Report. The remainder of the test was deferred with a required completion date of the first refueling.

The A&B Cleanup Ion Exchangers, A&B Strainers, and Cleanup Filter were verified to operate satisfactorily during testing completed by April 12, 1988. The A&B Cleanup Ion Exchangers exhibited low differential pressure operational data which was evaluated by Engineering and found to be acceptable. High differential pressure across the A&B Strainers also needed evaluated by Engineering. The "B" Strainer was determined acceptable because it was only slightly high, but the "A" Strainer required removal and inspection. No signs of clogging were observed and the "A" Strainer was retested with acceptable results. The Cleanup Filter exhibited a differential pressure reading high out of specification. Evaluation by Engineering determined that operation at this differential pressure reading was acceptable.

Testing of the "B" Steam Generator Blowdown Evaporator had not yet been completed during the time period covered by this supplement report. Test results will be discussed in another supplement report upon completion of remaining testing and resolution of any deficiencies.

2.11 Plant Communications Test (SOV-2.40A.01)

The Fuel Load Dedicated Calibration Jack System was satisfactorily completed prior to core load. Operability of the Communications System was proven prior to core load with the remainder of the test deferred until prior to Commercial Operations. The Intra Plant Page/Party System, audibility of the "Standby" and "Evacuation" alarms, and Plant Telephone System (PAX lines) were tested prior to Commercial Operations with several test deficiencies. Several broken telephones required repair. Background noise levels and evacuation alarm decibel levels throughout the plant did not meet the required acceptance criteria. Engineering was given the data and requested to evaluate and resolve the audibility problems. Completion of the test was then deferred until prior to December 31, 1987, and subsequently to the first refueling.

"Evacuation" and "Standby" alarms audibility was verified at various locations outside the Reactor Containment Building. Measured background sound levels and Evacuation Alarm sound levels did not meet acceptance criteria for several speaker locations in the Reactor Containment Building. Background sound levels (in absolute decibels or dba) were required to be less than the estimated maximum sound levels (given in NRC Interrogatory

Q 430.56-3). Evacuation Alarm sound levels were required to be 10 db above the estimated maximum sound levels in dba but not less than 75 dba at each tested plant location (based on ANSI/ANS N2.3-1979). Engineering evaluated the audibility problems. Their response agreed with the proposal to use actual background sound levels obtained instead of the estimated maximum sound levels as a basis for evaluating an alarm's sound level acceptability. Additionally, the responses stated that acceptance criteria requiring alarm sound levels 10 db above background sound be used only as a guideline and final acceptance should be based on alarms being clearly audible above background noise so as to initiate evacuation. Additionally, Engineering made recommendations for improving test results (i.e., adjusting speaker amplifiers to maximize volume, correcting speaker misalignments and providing instructions on sound level measurement locations for uniformity of data). Maintenance Work Requests were issued identifying speakers in the Reactor Containment Building requiring adjustment/repair and follow-up audibility tests using approved station procedures. Those speakers located outside Containment requiring adjustment, repair, and/or reorientation were retested at 100% power with acceptable audibility readings. Those areas outside Containment that showed a significant background noise level increase at 100% power from previous recorded levels at lower power levels, were verified acceptable during testing completed on March 16, 1988.

Proper operation of the Intra-Plant Page/Party System by paging and conversing with the Control Room from every page/party handset location utilizing all five channels was successfully completed on April 13, 1988.

Proper operation of the Plant Telephone (PAX) System, radio handsets at the Auxiliary Communications Panel, and special private telephone lines were verified prior to Commericial Operations with several test deficiencies. At the time of this supplement report, three (3) deficiencies had not yet been resolved, but were transferred to the Station's Communications Department for troubleshooting and testing. The deficiencies included: (1) installation, repair, or replacement of any PAX telephone extensions which could not be satisfactorily tested; (2) complete installation of the radio handsets #1 and #2 transmit/receive cables to the BVPS-1 tie in point; and (3) troubleshoot, repair, and test the Emergency Airlock private line phone. Following resolution of the above deficiencies, retesting will be completed by approved station procedures.

2.12 Non-QA Category I Heat Tracing Sytem Test (SOV-2.45D.01)

Testing of control logic and the ability to maintain system fluid temperatures within design specifications for the following systems was completed following Commercial Operations:

Quench Spray System
Water Treatment Demineralizer System
Condensate System

Steam Generator Blowdown System
Solid Waste Disposal System
Chemical & Volume Control System
Gaseous Waste Disposal System
Containment Vacuum System
Reactor Coolant System

This test had been deferred several times due to incomplete testing and deficiencies with the system. It was completed on March 24, 1988.

Proper operation of the various heat tracing indicators and alarms was satisfactorily verified. The Non-QA Category I freeze protection circuits were satisfactorily tested, but several circuits did not meet the test acceptance criteria. The deficient data was forwarded to Engineering for evaluation. The final result of the evaluation was that valid acceptance criteria for the freeze protection circuits could not be generated for testing conducted at other than the design conditions of -20F ambient with 58 mph winds. These conditions obviously could not be met for testing the system piping. Engineering stated that as problems were identified, an evaluation of the affected circuits would be made and appropriate corrective actions would be implemented. These actions will be covered by appropriate station procedures as they arise. Various heat tracing circuits that are designed to maintain process fluids at temperatures that will prevent precipitation of dissolved chemicals or thermal stress to components were satisfactorily tested, but several circuits did not meet the test acceptance criteria. Engineering was requested to perform an evaluation of the data and to investigate the possibility of relaxing the acceptance criteria values. Engineering stated that the identified deficient circuits would receive further evaluation for corrective action. Any modifications as a result of this evaluation will be implemented using existing station procedures.

During the 24 hour performance measurements, several circuits were determined to be acceptable even though some of the data fell below the established setpoint. Possible reasons for the low readings include changes in ambient or process temperatures, or a temperature lag between the time that the circuit energizes and the fluid begins to heat. These circuits were determined to be acceptable based on the ability of the circuit to reach the test temperature and the reheat to acceptable temperatures following readings that were below the setpoint.

This test was considered satisfactorily completed. However, as noted in the above discussion, there are a number of circuits that did not meet the stated acceptance criteria. An Engineering evaluation of problem circuits will continue to be made as these circuits are identified. Appropriate rework will be developed and completed using appropriate station procedures. During the 1987/88 winter season, only one line in the Non-QA Category I Heat Tracing System actually froze. This was a sensing line for a level transmitter on the Demineralized Water Storage Tank.

Temporary heaters were used to prevent further freezing on this circuit.

There were several QA Category I Heat Tracing lines that also froze during the winter season. Temporary supplementary heat tracing and insulation were added to these lines to prevent further freezing. Engineering will evaluate the affected circuits and identify permanent rework using appropriate station procedures.

2.13 Loose Parts Monitoring System Test (SOV-2.45A.01)

The Loose Parts Monitoring System provides monitoring of major reactor coolant system components in which a loose part could be trapped. The Loose Parts Monitoring System (LPMS) detects unusual noises that may be an indication of a loose metallic part. This is accomplished by the use of permanently mounted accelerometers which are on vital pieces of equipment that are located inside the Reactor Containment and inaccessible during plant operation.

This test was completed prior to Commercial Operations. It obtained Reactor Coolant System baseline (signature) data prior to core load, during Modes 5, 3, and 2, and at power levels of 5, 25, 50, 75, and 100% power, and verified the proper operation and ability of the Loose Parts Monitoring System to detect loose metal parts in the Reactor Coolant System. A detailed discussion on the test can be found in Section 7.6.10 of the original Startup Report. The LPMS vendor, Babcock & Wilcox Company (B&W), was forwarded the recorded data obtained during startup.

Subsequent to Commercial Operations, the report issued by B&W following the data analysis recommended several changes to system setpoints. It also recommended additional troubleshooting to determine the cause of the noise signal on the steam generator channels as discussed in the original Startup Report (Section 7.6.10). The data analysis showed that the noise was present on all steam generator channels, though it had a greater amplitude on steam generators A and B. The report further stated that if the noise was found to be flow induced, filter changes should be made to the signal processing cards for the affected channels.

During a short maintenance outage at Unit 2, a B&W representative was on site to assist in the recommended troubleshooting. After an inspection and impact testing of one of the system's accelerometers, it was believed that the noise problem was flow induced. The recommended filter changes were made, along with various adjustments to the system settings. In addition, an attempt was made to repair a reported problem with the data processing portion of the system that had developed after the completion of the data taking portion of the test. The microprocessor unit would lock up or send continuous erroneous data to the internal printer when multiple alarms were received. Following return to power operation, checks were made to determine the effectiveness of the work that had been performed on the Loose Parts Monitoring System. The filter changes and system setting changes resulted in better operation of the alarm detection circuits. However, the data processing portion of the system still exhibited the same problem when multiple alarms were received. Additional data was later taken to attempt to isolate the cause of the steam generator noise signal and to determine if there was any adverse impact to the plant because of the signal. Data was taken on steam and feedwater piping using vibration monitoring equipment. A tone at the turbine was investigated using a microphone connected to a vibration analyzer. Additional data was taken on the LPMS system at zero power and at various power levels during subsequent plant start-up. The LPMS channels were monitored while blowdown was isolated for the A steam generator. Finally, data was obtained from the Reactor Coolant Pump vibration monitors. The only conclusions from the accumulated data and weekly monitoring of the noise signal were as follows: The signal is of low amplitude until above 85% power level. Between 85 and 100% power the signal increases by approximately 400%, where it remains relatively constant. The signal appears to originate in or near the steam generators since it is not apparent at other equipment/piping that was monitored.

The requirements of the test procedure have been satisfactorily met. In respect to the noise signal observed on the steam generator channels, calculations show that the displacement necessary to produce the tone at the acceleration level observed is negligible, thus no further action is required. The continuing problems with the data processing portion of the system is a maintenance item that will be cleared using normal plant procedures. Since the cause of the malfunctions, per B&W, may be related to heat buildup in the LPMS cabinet, request was made to consider providing ventilation for the cabinet. Any action taken on that request will be handled by existing plant procedures.