U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-265/85023

Docket No. 50-265 License No. DPR-30

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Facility Name: Quad Cities Nuclear Power Station, Unit 2

Inspection At: Cordova, Illinois

Inspection Conducted: July 8 through 19, 1985

Inspectors: P. R. Rescheske for M. A. Ring — 8/13/85 Date

S. G. DuPont for M. A. Ring — 8/13/85 Date

Approved By: M. A. Ring, Chief
Test Programs Section — 8/13/85 Date

Inspection Summary

Inspection on July 8 through 19, 1985 (Report No. 50-265/85023(DRS))
Areas Inspected: Routine, unannounced safety inspection of surveillance of core power distribution limits, calibration of the local power range monitoring system, APRM calibration and core thermal power evaluation, determination of reactor shutdown margin, control rod drive performance testing, and additional startup testing. The inspection involved a total of 84 inspector-hours onsite by two NRC inspectors including 4 inspector-hours onsite during off-shifts.
Results: Of the six areas inspected, no violations or deviations were identified in three areas, and examples of one violation were identified in the remaining three areas (inappropriate procedures - Paragraphs 4.a, 4.b., 6.a, 7.b, and 7.c)

DETAILS

1. Persons Contacted

*T. Tamlyn, Services Superintendent
*R. Bax, Production Superintendent
*G. Spedl, Assistant Superintendent Technical Services
H. Lihou, Technical Staff Supervisor
*P. Knoespel, Lead Nuclear Engineer
E. Weinfurter, Nuclear Engineer
*B. Strub, Regulatory Assurance Administrator
*C. Norton, Quality Assurance Engineer

The inspectors also interviewed other licensee employees including members of the technical and operating staff.

*Denotes persons attending the exit meeting on July 19, 1985.

2. Surveillance of Core Power Distribution Limits

The inspectors reviewed a number of the licensee's surveillances of core thermal limits including the computer printouts of P-1 and OD-6. Procedure QTS 1519-5, "Nuclear Engineer's Surveillance During Unit Operation," Revision 1, is used by the licensee to periodically check plant parameters and acquire data during power operation (above 25% rated core thermal power). The inspector reviewed a sample of the checklists QTS 1519-S4 (Revision 5) completed between June 11 and July 16, 1985, and verified that the data was properly recorded and reviewed at the recommended frequency of at least three times per week.

The licensee performs and documents daily surveillances when core thermal power is greater than 25% of rated using the procedure QOS 005-3, "Unit Operator's Daily Surveillance of Nuclear Limits," Revision 3, and the checklist QOS 005-S1 (Revision 31). The inspector reviewed a sample of the surveillances completed in June 1985 and verified that the results and the frequency of testing satisfied Technical Specifications, and that the data was properly recorded and reviewed. The inspector determined that the following required surveillances were acceptable and that proper corrective actions were initiated when necessary:

- Core Thermal Power (CTP) not to exceed the rated reactor power (2511 MWt) as specified in the Unit 2 License DPR-30.

- Maximum Fraction of Limiting Power Density (MFLPD) is less than 1.00, which assures that all local Linear Heat Generation Rates (LHGR) are less than the Technical Specification 3.5.J limit of 13.4 kw/ft.

- MFLPD is less than Fraction of Rated Power (FRP), which assures compliance with Technical Specifications 2.1.A.1 and 2.1.B. When MFLPD exceeded FRP, appropriate corrective actions were initiated by the licensee (e.g., increasing APRM gains or lowering APRM trip settings).

- Maximum MAPRAT is less than 1.00, where MAPRAT is the ratio of the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) to the MAPLHGR limit. This ensures that Technical Specification 3.5.I is satisfied.

- Maximum Fraction of Limiting Critical Power Ratio (MFLCPR) multiplied by the CPR flow correction factor is less than 1.00, which assures compliance with Technical Specification 3.5.K and ensures that the Minimum Critical Power Ratio (MCPR) is maintained greater than that specified in Technical Specification 1.1.A.

- CPR correction for a Limiting Control Rod Pattern (LCRP) multiplied by the flow corrected MFLCPR is less than 1.00, which assures that a LCRP does not exist. When a LCRP apparently existed, the licensee verified that both Rod Block Monitors were operable per Technical Specification 4.3.B.5.

No violations or deviations were identified.

3. Calibration of the Local Power Range Monitoring System

The inspector reviewed the licensee's procedures and test results for the calibration of the Local Power Range Monitor (LPRM) system and verified that the calibrations are performed at the required frequency of 1000 effective full power hours as specified in Table 4.1-2 of the Technical Specifications. The following observations were made during this review:

a. Procedure QTS 1311-1, "Full Core LPRM Calibration," Revision 8, was required for the initial LPRM calibration following the refueling outage. The inspector reviewed the computer printouts (P-1, OD-1, OD-8, and the prime computer OD-5) and the Traversing Incore Probe (TIP) traces from the calibration performed on June 9, 1985, and verified that the appropriate current adjustments were made and that the data package was properly reviewed (cover sheet QTS 1311-S1). In addition, the inspector determined from the data that all newly installed LPRMs were adjusted to 700 micro amps as required prior to startup.

b. Procedure QTS 1311-3, "Individual LPRM Detector Calibration," Revision 2, was performed on June 20, 1985, to calibrate selected LPRMs. The inspector reviewed the computer printouts, TIP traces, data sheet QTS 1311-S6 (Revision 2), and the LPRM Calibration Data printouts, and verified that the calibration was properly

documented and reviewed (coversheet QTS 1311-S1). In addition, the inspector verified that the APRMs were adjusted (form QTS 1512-S2) subsequent to recalibrating the LPRMs.

c. Procedure QTS 1311-4, "Bypassing LPRMs," Revision 3, is used by the licensee to place LPRM detectors in bypass and then return them to service following recalibration. The inspector noted that three LPRMs had been bypassed on June 8, 1985, and verified that this was properly documented on the checklists QTS 1311-S7 (Revision 2). In addition, the inspector verified that APRM operability was satisfied (Technical Specification Table 3.1-3) and that the APRMs were calibrated after the LPRM was bypassed. The inspector observed, however, that the procedure did not include instructions for returning the LPRM to service. The licensee agreed to revise the procedure to include the method for returning the LPRMs to service after recalibration.

During the review of the above procedures, the inspector observed that some references to Technical Specifications and checklists were missing. Also, certain information required by the procedure was not being obtained during the calibration because it was either obsolete (i.e., computer paper tape) or not necessary (i.e., Prime computer OD-5 obtained rather than the process computer OD-3). The licensee agreed to modify the procedures to reflect the current methods used at the facility. The inspector has no further concerns regarding these procedures.

No violations or deviations were identified.

4. APRM Calibration and Core Thermal Power Evaluation

The inspector reviewed the licensee's procedures and test results for heat balance calculations and Average Power Range Monitor (APRM) calibrations. The following observations were made during this review:

a. Procedure QTS 1512-1, "Nuclear Engineer's Method for APRM Calibration," Revision 7, outlines the methods used by the Nuclear Engineers to calibrate the APRMs at both high and low powers and to perform reactor heat balances. At low power, during the startup after the refueling outage, the APRMs were initially calibrated based on the percent bypass valves open. The inspector reviewed the low power APRM calibration performed on June 6, 1985, at 10% CTP, and verified that the data was properly recorded on form QTS 1512-S1 (Revision 2) and that the adjusted APRM readings were within 2% of the actual CTP. According to 10 CFR 50, Appendix B, Criterion V, as implemented by the CECo Quality Assurance Manual (Q.R. No. 5.0 and Q.P. No. 5-51), activities affecting quality shall be prescribed

by appropriate written procedures. Contrary to this, the inspector observed that form QTS 1512-S1 was inappropriate for performing the test in that it did not indicate the persons who performed the test or specify the date and Unit (1 or 2) on which the test was performed. This is considered an example of a violation (265/85023-01a(DRS)).

The licensee performed heat balance calculations at high power on June 7, 1985, (at 25% CTP) and June 14, 1985, (at 100% CTP) to verify the accuracy of the process computer's OD-3 program. The inspector reviewed the OD-3 printouts, the Prime computer OD-5 printouts, form QTS 1512-S5, the heat balance calculations performed using the backup Prime computer POWER program (form QTS 1512-S3), and the subsequent APRM calibrations documented on form QTS 1512-S2. The inspector verified that the data was properly recorded and reviewed, that the OD-3 and OD-5 edits were in agreement with the POWER heat balance calculation, and that the adjusted APRM readings were within 2% of the actual CTP. The inspector noted that the procedure required OD-7 and OD-8 printouts to be obtained; however, the licensee had omitted these printouts because they were unnecessary for performing the test. The licensee plans to delete this requirement from the procedure.

During power operation, the license performs APRM calibrations using form QTS 1512-S2 and the value for CTP from an OD-3 edit. The inspector reviewed a sample of calibrations completed in June 1985, and verified that the data was properly recorded and reviewed and that the adjusted APRM readings were within 2% of the actual CTP.

b. Surveillance procedure QOS 700-6, "APRM High Flux (Heat Balance) Calibration Test," Revision 6, outlines the methods used by the Operating staff to calibrate the APRMs by heat balance. The calibration is required once every seven days as specified in Table 4.1-2 of the Technical Specifications and the calculation is documented on data sheet DOS 700-S7. The inspector reviewed the calculations performed in June 1985, and verified that the data was properly recorded and reviewed at the required frequency and that the APRMs were recalibrated when the adjusted APRM readings deviated by greater than 2% from the actual CTP (the value for CTP is from the process computer OD-5 edit). Step F.2.b of the procedure outlines the method to be used if both the process computer and a Nuclear Engineer are not available. This hand calculation is documented on form QOS 700-S4 with the use of the nomograph QOS 700-S6. The inspector attempted to perform a heat balance calculation using this approved method with actual plant data, but was unsuccessful due to the inadequacies in the instructions, data sheet, and nomograph. According to 10 CFR 50, Appendix B, Criterion V, as implemented by the CECo Quality Assurance Manual (Q.R. No. 5.0 and Q.P. No. 5-51), activities affecting quality

shall be prescribed by appropriate written procedures. Contrary to this, the inspector found that the above method was inadequate for performing a heat balance calculation and obtaining reasonable results due to the complexity of the instructions and the inaccuracy of the nomograph. This is considered an example of a violation (265/85023-01b(DRS)). The inspector noted that this method is not the preferred method and has recently not been used; therefore, no significant safety issue exists. The licensee plans to delete this method from the procedure and rely on the other backup heat balance methods currently in use.

No additional violations or deviations were identified.

5. Determination of Reactor Shutdown Margin

The inspector reviewed procedure QTS 1104-1, "Shutdown Margin Subcritical Demonstration," Revision 8, used by the licensee to satisfy the Technical Specification surveillance requirement 4.3.A.1 subsequent to the Cycle 8 core load. The inspector also reviewed information contained in the Cycle 8 Startup Package provided by the General Electric Company. The licensee performed a local shutdown margin (SDM) demonstration on June 5, 1985, using the 2-rod diagonally adjacent subcritical method. The results of the test indicted that the reactor remained subcritical with the strongest worth control rod fully withdrawn and the diagonally adjacent control rod at notch 36. The inspector observed that with this rod pattern and the predictive data supplied by GE, the Technical Specification limit of 0.773%ΔK (at 68°F) was satisfied (NOTE: Technical Specification SDM = .25%ΔK + R + Settling Penalty, where R is 0.473%ΔK and the Settling Penalty is 0.05%ΔK). The inspector reviewed the test approval sheet QTS 1104-S3, the data sheets QTS 1104-S1 and QTS 1104-S2, and the control rod maneuver sheet QTP 1600-S3, and verified that the data and results were properly recorded and reviewed. In addition, the inspector verified that the procedure and the methodology was technically adequate and in accordance with the Technical Specifications.

No violations or deviations were identified.

6. Control Rod Drive Performance Testing

The inspector reviewed the licensee's procedures and test results for startup testing of the control rod drives. The following observations were made during this review:

a. Procedure QTS 130-1, "Control Rod Timing and Position Indication Check," Revision 7, was used by the licensee to satisfy Technical Specification requirement 3.3.A.3 (rod position indication) and to adjust, if necessary, the insertion and withdrawal times of each control rod drive after final loading of a control cell. The inspector reviewed the data obtained between May 1 and June 3,

1985, and noted that the verification of rod position indication, as required by Step F.3 of the procedure, was not recorded on data sheet QTS 130-S1 for any of the rods tested. According to 10 CFR 50, Appendix B, Criterion V, as implemented by the CECo Quality Assurance Manual (Q.R. No. 5.0 and Q.P. No. 5-51), activities affecting quality shall be prescribed by appropriate written procedures. Contrary to the above, procedure QTS 130-1 was inadequate in that it did not clearly state which data was to be recorded in the available columns on data sheet QTS 130-S1; hence, the required verification of rod position indication was not documented. This is considered an example of a violation (265/85023-01c(DRS)). In addition, the inspector observed the following weaknesses with procedure QTS 130-1 and data sheet QTS 130-S1:

(1) Data was obtained by a number of licensee personnel between May 1 and June 3, 1985, but the data sheet does not indicate who recorded the data or when the data was obtained. As an example, data for rod L-2 was obtained over a one-month period; if a problem with the data existed and a resolution was necessary, then knowledge from the individual persons performing the test would be required.

(2) Step F.4.h of the procedure requires that steps b through h be repeated until the rod establishes consistent speeds within the specified time band (48-60 seconds) for insert and withdrawal. Step F.4.h is not indicative of the test actually performed in that it implies that if either withdrawal or insert speeds are adjusted, then both withdrawal and insert times are to be verified. However, because the insert and withdrawal times are independent of each other, the licensee only verifies the speed that was adjusted. The licensee plans to revise the procedure to reflect the actual test to be performed.

(3) Several rods, including rod L-2, did not meet the requirement for withdrawal times of between 48 and 60 seconds. However, these rods were accepted by the licensee after the rods successfully passed the rod notch out test. This is another example of the inadequacy of the procedure because the rod notch out test is not prescribed as an option to the acceptance criteria (48 to 60 seconds). The inspector noted that the rod notch out test is a technically adequate alternative. The licensee indicated that they had realized that the procedure needed to be revised to include the rod notch out test as part of the acceptance criteria, but had not as yet initiated the revision.

(4) The inspector also observed that excessive write-overs appear on the data sheets and that some of the data was not legible.

For each of the above perceived weaknesses, the licensee has already commenced appropriate corrective actions and the inspectors have no further concerns.

b. Procedure QOS 300-4, "Control Rod Coupling Integrity, Neutron Instrumentation Response," Revision 2, was used by the licensee to satisfy Technical Specification requirements 3.3.B.1 and 4.3.B.1.a. The inspector reviewed the data obtained on June 5, 1985 and noted that the results indicated that rod coupling integrity was verified. The inspector observed that excessive write-overs appear on data sheet QOS 300-S1 and that some of the data was not legible. The licensee acknowledged the inspector's concern.

c. Procedure QTS 130-3, "Control Rod Friction and Settle Testing," Revision 8, was used by the licensee to determine if core component interference existed with control rod travel. The inspector reviewed the data obtained on May 3, 1985, and found the results to be adequate. The inspector observed that the instructions for operating the differential pressure (DP) cell were within the section of the procedure pertaining to the optional Textronix Storage Oscilloscope method. The licensee normally uses the strip chart recorder to perform the test, but the DP cell operation is not referenced in this section of the procedure. The licensee agreed to revise the procedure to include reference to DP cell operation in the correct sections of the procedure.

d. The inspector reviewed the data obtained by the test procedures QOS 300-6, "Control Rod Coupling Integrity Overtravel Check," and QTS 130-4, "Control Rod Scram Timing," and verified that the data was properly recorded and reviewed, and that the Technical Specifications requirements were satisfied.

No additional violations or deviations were identified.

7. Additional Startup Testing

The inspector reviewed Special Test No. 2-50, "Quad Cities, Unit 2, Cycle 8, Startup Test Program," which outlines the method used by the licensee to perform the required startup tests subsequent to the refueling outage. The inspector verified that the test was approved by the appropriate personnel and that a safety evaluation was performed prior to use. This procedure references the tests that should be completed during startup, including those described above in Paragraph 2 through 6. The inspector reviewed the following tests and found them acceptable unless otherwise noted:

a. Procedure QTP 1113-1, "Process Computer Nuclear Program Check," Revision 6, specifies a method for determining the acceptability of the process computer for reactor core nuclear calculations. Prior to startup, the computer was updated with the new constants for Cycle 8 and then verified correct on checklist QTP 1113-S1. During power operation certain checks of the computer update were completed; i.e., comparison of the process computer heat balance (OD-3) to a hand heat balance calculation (Prime backup computer program POWER). Procedure QTS 1519-4, "Calculation of Thermal Limits Using the Prime Computer," Revision 6, was used by the licensee to compare the P-1 and OD-6 edits from the Prime with those obtained from the process computer to check for calculational accuracy. Data sheets QTS 1519-S2 and QTS 1519-S3 were used to document this test. The inspector reviewed the procedures, computer printouts, and data sheets for the process computer checks, and verified that the data was properly recorded and reviewed and that the computer was approved for use during power operation.

b. Procedure QTP 1106-2, "Initial In-Sequence Criticality Estimate Evaluation," Revision 1, was used by the licensee to evaluate the acceptability of the analyzed critical control rod pattern for the initial startup subsequent to the refueling outage. The inspector reviewed the completed test performed on June 5, 1985, and verified that the results indicated that the Technical Specification requirement 3.3.E was satisfied. This requirement states that the reactivity difference between the actual critical rod configuration and the expected configuration shall not exceed 1.0%ΔK. The actual critical data was recorded on form QTP 1106-S1 (Revision 3) and the evaluation of the estimated critical configuration was documented on form QTP 1106-S3 (Revision 1). The inspector noted that form QTP 1106-S3 only included the results from the calculation and did not specify the method used to calculate the reactivity difference. GE supplies the data necessary for this calculation in the Startup Package, but this information is provided in the form of tables and graphs. A number of methods exist (some more accurate than others) for calculating the actual reactivity from the critical rod pattern; i.e., estimating the value from the graphs and interpolating keff or ΔK values using the tables. The inspector also noted that form QTP 1106-S3 specified a calculation to be performed for the moderator temperature correction rather than referring to the graph provided by GE. According to 10 CFR 50, Appendix B, Criterion V, as implemented by the CECo Quality Assurance Manual (Q.R. No. 5.0 and Q.P. No. 5-51), activities affecting quality shall be prescribed by appropriate written procedures. Contrary to this, the inspector found that the above procedure was inappropriate in that an approved method was not specified for the initial criticality evaluation. This is considered an example of a violation (265/85023-01d(DRS)). The inspector performed calculations using the different methods

and observed that the results were similar and that they satisfied the acceptance criteria; therefore, no significant safety issue exists. The licensee plans to revise the data form to include the preferred method for the calculations. Specifying a method will provide consistent results from cycle to cycle and also allow the calculations to be more easily reproduced if necessary.

c. Procedure QTS 1300-1, "Operating Rod Inventory Comparison for Reactivity Anomaly Surveillance," Revision 7, is used by the licensee to satisfy the Technical Specification surveillance requirement 4.3.E. According to this requirement, a comparison of the critical rod pattern to the expected, based on past data, shall be performed at least every equivalent full power month. The inspector reviewed the procedure, data sheet QTS 1300-S1, and the reactivity anomaly projection curve (provided by GE) for the tests performed on June 14 and July 12, 1985. The inspector verified that the data was properly recorded and reviewed, that the methodology was technically adequate, and that the results satisfied the acceptance criteria. The inspector also reviewed the Prime computer program ANOM used to determine the %ΔK and verified that the Cycle 8 constants were on file and that the program was approved for use. Steps F.5 through F.10 of the procedure are used if the Prime computer is unavailable and a hand calculation is necessary. The inspector verified that this hand calculation gave results consistent with those obtained from the Prime calculation but noted that a form did not exist to document the calculation. The licensee agreed to create a data sheet on which to record the calculation and results of the hand calculation. The inspector has no further concerns regarding this procedure.

d. Procedure QTP 1118-1, "Core Power Distribution Symmetry Analysis," Revision 5, outlines the methods used to determine the reproducibility of the TIP system readings and the magnitude of core power distribution asymmetries. The inspector reviewed the computer printouts and the TIP traces obtained from the test performed on June 20, 1985, and verified that the total TIP uncertainty was less than 9% and that the maximum deviation between symmetrically located TIP pairs was less than 25%.

e. Procedure QTP 1600-3, "Flow Control Line Determination," Revision 5, was used by the licensee to determine core power vs. flow characteristics near the 100% rod pattern line. Power and flow data was acquired on June 14, 1985, while decreasing load in 50 MWe increments from 100% CTP to 54% CTP. The inspector reviewed the data sheet QTP 1600-S8 (Revision 1) and verified that the data was properly recorded. According to 10 CFR 50, Appendix B, Criterion V, as implemented by the CECo Quality Assurance Manual (Q.R. No. 5.0 and Q.P. No. 5-51), activities affecting quality shall be prescribed by appropriate written procedures. Contrary to this, the inspector

observed that data sheet QTP 1600-S8 was inappropriate for performing the test in that it did not indicate the persons who performed the test or specify the date and Unit (1 or 2) on which the test was performed. This is considered an example of a violation (265/85023-01e(DRS)).

f. Procedure QTP 1130-1, "Core Flow Calibration," Revision 4, was used by the licensee during startup to determine reactor core flow, to calibrate the total and loop jet pump flow instrumentation, and to verify jet pump flow consistency. The inspector reviewed the procedure and the completed data sheet QTP 1130-S1, and verified that the data was properly recorded and reviewed, that the results met the acceptance criteria, and that the test equipment used to collect the data was calibrated and traceable to national standards.

g. Procedure QTP 1130-2, "Jet Pump Base Data," Revision 4, was used by the licensee during startup to obtain baseline data for jet pump operability and integrity surveillance throughout Cycle 8. The inspector reviewed the data obtained on data sheets QTP 1130-S4 and QTP 1130-S5 for the test completed on June 12, 1985. The data sheets are used to record data at intervals of approximately 5% CTP. The inspector observed that at approximately 95% CTP, two points of data were not recorded on data sheet QTP 1130-S4: the differential pressure across the core plate (DPC at panel 902-5 recorder) and the reactor vessel differential pressure (meter indication at panel 902-3). Both of these data points are manual backups for the process computer OD-5 edit. The inspector identified this as an isolated case of a weak review of test data by the licensee, and noted that during the final review of the startup test program, the licensee would have detected the missed data points. The inspector has no further concerns regarding this item.

h. Procedure QTP 1109-1, "SRM Performance Check," Revision 2, was used by the licensee during startup to verify that the Source Range Monitors (SRMs) can detect neutron flux and to verify SRM and Intermediate Range Monitor (IRM) overlap. The inspector reviewed the data obtained on June 5, 1985, and verified that the data indicated an adequate signal-to-noise ratio for SRM operability and sufficient overlap between SRMs and IRMs. The inspector noted, however, that several entires on data sheet QTP 1109-S2, "SRM Performance - Response to Rod Movement," were marked over and not legible. The licensee acknowledged the inspector's concern.

i. Procedure QTP 1110-1, "IRM Performance Check," Revision 2, was used by the licensee during startup to verify that the IRMs can detect neutron flux and to verify IRM overlap. The inspector reviewed the data obtained on June 5, 1985, and verified that the data indicated that the IRMs adequately detected continuous neutron flux

throughout the IRM range, including continuity through ranges six and seven. In addition, the inspector verified that the overlap was adequately demonstrated. The inspector noted that Step F.2 in the procedure references data sheet QTS 1512-S1, "Low Power APRM Calibration," as an option for the IRM to power relationship if an accurate heat balance is not available. However, data sheet 1512-S1 is not referred to in either the Checklist section or the Reference section of the procedure. The license acknowledged the inspector's concern and agreed to revise the procedure to include proper references to QTS 1512-S1.

No additional violations or deviations were identified.

8. Exit Interview

The inspectors met with the licensee representatives (denoted in Paragraph 1) on July 19, 1985. The inspectors summarized the scope and findings of the inspection. The licensee acknowledged the statements made by the inspectors with respect to the violation. The inspectors also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspectors during the inspection. The licensee did not identify any such documents/processes as proprietary.