U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-346/85009(DRS)

Docket No. 50-346 License No. NPF-3

Licensee: Toledo Edison Company
Edison Plaza, 300 Madison Avenue
Toledo, OH 43652

Facility Name: Davis-Besse, Unit 1

Inspection At: Oak Harbor, OH

Inspection Conducted: April 8-26, 1985

Inspectors: M. L. McCormick-Barger — 5-28-85 Date

M. A. Ring — 5-28-85 Date

M. J. Farber — 5-28-85 Date

Approved By: R. D. Lanksbury, Acting Chief
Test Programs Section — 5-28-85 Date

Inspection Summary

Inspection on April 8-26, 1985 (Report No. 50-346/85009(DRS))
Areas Inspected: Routine, announced inspection of control rod worth measurements, shutdown margin calculations, power doppler coefficient measurement, moderator temperature coefficient measurement, thermal power evaluation, flux/delta flux/flow reactor trip, core power distribution, incore-excore calibration and estimated critical boron concentration calculations. The inspection involved 166 inspector-hours onsite by three inspectors including six inspector-hours onsite during off-shifts.
Results: Of the nine areas inspected, no items of noncompliance or deviations were identified in five areas; two items of noncompliance were identified in the four remaining areas (failure to perform a procedure, Paragraph 6.b.; failure to adequately evaluate test results, Paragraphs 2.a., 2.b.(1), 5.a., and 8.a.).

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DETAILS

1. Persons Contacted

   *T. Murray, Assistant Vice President of Nuclear Operations
   +*J. Lingenfelter, Technical Superintendent
   *D. Dibert, Nuclear and Performance Engineer - Technical Section
   *E. Caba, Station Performance Engineer
   *B. Sarsour, Senior Assistant Engineer/Performance
   C. Berger, Senior Assistant Engineer
   L. Simon, Operations Supervisor
   J. DeSando, Instrument and Control Department
   T. Eisley, Instrument and Control Department
   K. Yarger, Instrument and Control Department
   *S. Wideman, Licensing
   *D. Stephenson, Licensing Department
   *J. Byrne, Quality Assurance Senior Auditor

   *Denotes those personnel present at the exit interview on April 26, 1985.
   +Denotes those personnel present at an exit interview on May 1, 1985, conducted via telephone conversation.

2. Control Rod Worth Measurements

   The inspector reviewed the following enclosures to Surveillance Test ST 5010.03, Revision 6, "Post Refueling Physics Testing," for Fuel Cycle 5 for technical adequacy and verified that the results satisfied the acceptance criteria:

   - Enclosure 6 (Step 6.7), "All Rods Out Critical Boron Concentration"
   - Enclosure 9 (Step 6.10), "Deboration to Zero Rod Index"
   - Enclosure 11 (Steps 6.12 and 6.13), "Ejected Rod Worth Measurement"
   - Enclosure 12 (Step 6.15), "Boration of Rods to Operating Position"
   - Enclosure 13 (Step 6.18), "Calculation of Differential Boron Worth"
   - Enclosure 19, "Rod Worth Data Analysis"

   The inspector had the following comments with respect to:

   a. Enclosure 9 (Step 6.10)

      The inspector reviewed the calculations for Group 7 integrated regulating rod worth and noted that the value given in Enclosure 9, Data Sheet 2, for the rod worth of Group 7 from 95% WD (withdrawn) to 100% WD was 0.052% $\Delta K/K$. This value was indicated as being taken from Step 6.7. However, the data in Step 6.7 indicated that 0.052% $\Delta K/K$ was the worth of Group 7 from 93% WD to 100% WD. As a result,

the worth of Group 7 from 93% WD to 95% WD was added twice in the calculation of the integral control rod worth for Group 7. This is considered to be a violation of 10 CFR 50, Appendix B, Criterion XI and an example of an item of noncompliance (346/85009-01a(DRS)). The licensee demonstrated that the result of the corrected calculation of Group 7 integral rod worth was still within the acceptance criterion. In addition, after the date at which ST 5010.03 was completed and prior to the beginning of this inspection, the licensee had instituted measures requiring independent review of test results. Therefore, no further response is required.

b. Enclosure 11 (Steps 6.12 and 6.13)

(1) The inspector reviewed the calculations related to the ejected rod worth measurement and noted that the value given as the regulating rod worth at 58% WD was 0.540% $\Delta K/K$ whereas, from the data in Step 6.10 the correct value was 0.503% $\Delta K/K$. Therefore, the licensee calculated a value of 0.569% $\Delta K/K$ for the Measured Ejected Rod Worth Using Rod Swap, whereas this value should have been 0.606% $\Delta K/K$. Based on the above, the licensee performed an incorrect calculation for ejected rod worth which is considered to be a violation of 10 CFR 50, Appendix B, Criterion XI and an example of an item of noncompliance (346/85009-01b(DRS)). The licensee has subsequently reperformed the calculations using the correct values and instituted measures for independent review of results and, therefore, no further response is required.

(2) Although the initial rod position was established correctly for the ejected rod worth measurement, the basis used to establish this rod position was not identified in the procedure. The licensee intends to modify the procedure to identify acceptable initial conditions for rod position. This is an open item pending procedure revision and subsequent NRC review (346/85009-02(DRS)).

No additional items of noncompliance or deviations were identified.

3. Shutdown Margin Calculations

The inspector reviewed the following procedures for technical adequacy related to shutdown margin calculations and against the acceptance criteria and frequency of performance requirements of Technical Specification 4.1.1.1.1, "Boration Control - Shutdown Margin." The inspector had no concerns related to this review.

System Procedure SP 1103.15, Revision 5, "Reactivity Balance Calculations"

Surveillance Test ST 5010.03, Revision 6, "Post Refueling Physics Testing," (Enclosure 14 (Step 6.19), "Calculation of Shutdown Margin")

Surveillance Test ST 5099.01, "Miscellaneous Instrument Shift Check"

Surveillance Test ST 5099.02, "Miscellaneous Instrument Daily Checks"

No items of noncompliance or deviations were identified.

4. Power Doppler Coefficient Measurement

The inspectors reviewed PT 5175.03, "Power Doppler Coefficient," Revision 2, performed on February 13, 1985, in order to examine the licensee's procedure for measuring the power coefficient of reactivity and the results of the most recent measurement. The inspectors verified that prerequisites, precautions, and plant conditions were met, calculations were correct, values obtained were within acceptance criteria, and results were reviewed in accordance with administrative requirements.

No items of noncompliance or deviations were identified.

5. Moderator Temperature Coefficient Measurement

The inspectors reviewed ST 5010.03, "Post Refueling Physics Testing," Revision 6, Enclosure 7, and ST 5010.02, "Moderator Temperature Coefficient Measurement," Revision 6, in order to examine the adequacy of the licensee's procedure for measuring the temperature coefficient of reactivity and the results of the most recent measurements. The following items were noted during the review.

a. The inspector attempted to check the calculations performed for ST 5010.02, "Moderator Temperature Coefficient Measurement," (at power) but received a different result from the licensee. The licensee calculated the reactivity change for the second measurement of the temperature coefficient to be $-0.07198\% \Delta K/K/^{o}F$ while the inspector (using the licensee's numbers) calculated the same reactivity change to be $-0.07398\% \Delta K/K/^{o}F$. Further, for the same second measurement, the licensee utilized an incorrect value for xenon (Xe_{21}) of $-2.585\% \Delta K/K$. From the numbers recorded on page 9, the correct value should be $-2.576\% \Delta K/K$. Based on the above items, the licensee performed an incorrect calculation for moderator temperature coefficient which is considered to be a violation of 10 CFR 50, Appendix B, Criterion XI, and an example of an item of noncompliance (346/85009-01c(DRS)). This incorrect value was also used in PT 5175.03, "Power Doppler Coefficient." The licensee has subsequently reperformed the calculations using the correct values and instituted measures for independent review of results and, therefore, no further response is required.

b. ST 5010.03, Enclosure 7, regarding measurement of the temperature coefficient was performed with control rod group 7 90% withdrawn. The results, however, were compared to a Physics Test Manual table which is based on control rod groups 1 through 7 at 100% withdrawn. While the magnitude of the difference is small, the licensee should either perform testing at the conditions described

for the acceptance criteria or address any difference in the evaluation of results. The licensee has subsequently performed an evaluation of this difference and included the evaluation in the test results package. The inspector has no further concerns in this area.

No additional items of noncompliance or deviations were identified.

6. Thermal Power Evaluation

The inspectors reviewed ST 5030.01, "RPS Daily Heat Balance Check," Revision 7, and PT 5131.02, "Verification of Computer Calculations," Revision 0, in order to assess the technical adequacy and check the evaluation of results. In addition, the inspectors reviewed the circumstances surrounding the exceeding of a Technical Specification Limiting Condition for Operation involving the determination of thermal power. The inspectors' findings are as follows:

a. The results associated with ST 5030.01 can be calculated either entirely by hand or by inputting values by hand into a RAMTEK computer which then performs the calculation and provides a printout. The input values required for these calculations are obtained from the plant's process computer which assigns a specific plant parameter output to a unique computer point. The inspectors noted that the hand calculation and the RAMTEK calculation data sheets for ST 5030.01 were filled out such that the labeling of the computer points did not always match the values used. Although, in some cases, the individual performing the surveillance test labeled the source of the data, in other cases this was not done and, for these cases, an independent reviewer is unable to determine which computer points were used in the calculations. Further, while the data sheet from ST 5030.01 records reactor coolant pumping power in megawatts, the licensee's designated reviewer noted that the input for the RAMTEK program requests these inputs in kilowatts. The individuals performing the heat balance, however, were simply transferring the numbers from the data sheet to the computer input without accounting for the megawatt/kilowatt difference. The licensee had recognized these difficulties with the heat balance determination and was in the process of a procedure or program revision to correct the difficulties. The inspector has no further concerns in this area.

b. Following the inspectors' request to review all of the data associated with heat balance determination of thermal power, the licensee acknowledged that PT 5131.02, which performs a manual check of computer heat balance, had not been performed during the initial startup and run for Cycle 5 which lasted from January 15, 1985, through March 21, 1985. PT 5131.02 states, "The frequency for performing this test shall be monthly at the beginning of a fuel cycle until the computer has been verified to consistently perform satisfactorily." Further, the Reload Report for Cycle 5 requests special emphasis be placed on startup testing and, in particular, a heat balance be performed as early as possible. While the licensee may have met the wording of the reload report by

performing the computer heat balance as early as possible, failure to perform the manual check of the computer heat balance did not appear to the inspectors to indicate special emphasis was placed on startup testing. Failure to perform PT 5131.02 when required is considered to be a violation of 10 CFR 50, Appendix B, Criterion V, and an item of noncompliance (346/85009-03(DRS)).

c. On April 14, 1985, the steam generator No. 1 feed flow transmitter (FTSP2B2) which indicates on computer point F673, was determined to have filled with water and failed. The transmitter was tagged out and a temporary change to Surveillance Test ST 5030.01 was written which deleted computer point F673 from the average of the four feed flow readings in calculating thermal power by heat balance. While the temporary change deleted F673 from the procedure for a manual heat balance, operations and Instrumentation and Calibration (I&C) personnel involved in the above actions apparently did not realize that the failed feed flow indicator still inputted to the computer calculation of thermal power by heat balance. The Engineering Department which was knowledgeable of the computer inputs was unaware that the flow transmitter was out of service. From April 16 until discovery that the computer heat balance was incorrect on April 20, 1985, the licensee relied primarily on the computer heat balance for determination of reactor thermal power. On April 17, 1985, the licensee did check the thermal power determination as a routine part of comparing feed flow, power by heat balance, and generated megawatts electric during a startup. At this time, however, power as determined by computer heat balance was 80% to 90%. Coincidentally, the failure of the feed flow transmitter was such that the transmitter output was indicating a feed flow equivalent to 80% to 90% power. The instrument is a Bailey transmitter with a -10 volts to +10 volts range. When the instrument fails, it fails to 0 volts. The instrument is not linear and calibration records from a calibration on January 11, 1985, were such that 0 volts should be equivalent to just under 80% power with a ±.5% of full scale tolerance. After the fact examination of the feed flow readouts indicate the transmitter for computer point F673 failed to an indicated 87% power. Since this was roughly equivalent to the actual power level on April 17, 1985, the error was masked from the checks that were made.

Subsequently, the licensee continued to increase power. As power increased further away from the fixed feed flow transmitter output of approximately 87% power, the computer heat balance calculation became biased low such that actual thermal power was higher than thermal power as determined by computer heat balance. During this time nuclear instrumentation (NIs) was adjusted to be equal to thermal power by computer heat balance within an allowable band as determined from Figure 1 of ST5030.01. This figure allows NIs to indicate higher than thermal power but not lower.

From 11:50 a.m. on April 19, 1985, to 2:50 a.m. on April 20, 1985, the licensee recorded reactor coolant flow rate values in the operator logs from 1.79% to 2.065% lower than the reactor coolant

flow rate limit of Technical Specification 3.2.5.. This Technical Specification requires that Reactor Thermal Power be limited by 2% from rated thermal power for each 1% that reactor coolant flow rate is less than the flow rate limit within 4 hours. Therefore, during the above period, reactor thermal power was required to be limited to less than 96.42% to 95.87%. At 5:18 p.m. on April 19, the licensee stopped increasing power at 95.5% as determined by computer heat balance in order to stay below the thermal power limit. However, the computer heat balance was incorrect in that it still utilized an input from the failed feed flow transmitter. The licensee maintained approximately this power level until 2:20 a.m. on April 20, 1985, when operations personnel realized electric output in megawatts (MWe) was too high for the indicated thermal power. Power level by core differential temperature indicated 99.2% and MWe showed 903 MWe which is about 98%. A manual calculation of thermal power was performed indicating an actual power of 98%. The licensee then reduced power to 95.4% by manual secondary heat balance calculation. The Limiting Condition for Operation (LCO) from Technical Specification 3.2.5 was therefore violated in that reactor thermal power was allowed to exceed the required limit by approximately 2%. This action is considered an unresolved item (346/85009-04(DRS)) pending further NRC evaluation of related circumstances and whether escalated enforcement action is appropriate.

In addition to immediate reduction in power, the licensee has also taken the following corrective actions to prevent a recurrence of exceeding the LCO.

- ST5030.01, "RPS Daily Heat Balance Check," has been modified to require operators to compare power by computer heat balance to other power indicators as a check of the calculation.

- PT5131.02, "Verification of Computer Calculation," has been modified to require a weekly manual calculation to compare with the computer heat balance and a check of the computer inputs as well.

- A special order, Generic Guidance Memorandum SPO-2, was written and approved on April 26, 1985, providing guidance to station personnel on the importance of the heat balance calculation and lists the computer points and individual instruments affecting the heat balance. The special order requires notification of the Technical Section prior to any maintenance on the instruments listed.

The inspector also noted that one of the key parameters involved in exceeding the LCO, reactor coolant flow rate, is believed by the licensee and the NSSS vendor, Babcock & Wilcox (B&W), to be an artificial indication of low reactor coolant flow rate. In the Reload Report for Cycle 5, the licensee and B&W describe how adding burnable poison rod assemblies (BPRAs) to previously unblocked channels causes less coolant flow to bypass the core and more flow through the core area. However, the increased resistance to flow

across the reactor produces a lower reactor coolant flow in the loops for the equivalent pumping power as compared to previous cycles. The licensee contends this lower loop flow is actually equivalent to full flow through the core due to more coolant passing through the core area. The licensee had previously submitted an amendment to the Technical Specifications to reflect this fact. Based on the above discussion, the licensee believes that while the LCO on power level versus flow rate was actually exceeded, the safety impact was minimal in that the full core flow was present even though indicated loop flow was lower than the reactor coolant flow rate limit of Technical Specification 3.2.5.

Further actions regarding the above issue are expected to be addressed in a subsequent Inspection Report.

No additional items of noncompliance or deviations were identified.

7. <u>Operational Event Follow-up - Flux/Delta Flux/Flow Reactor Trip</u>

At 3:53 a.m. on April 24, 1985, while operating at 95% power as indicated by computer heat balance, the reactor tripped on flux/delta flux/flow. All four channels actuated within 15 milliseconds of each other. At the time the trip occurred the following conditions existed:

- Plant temperature, pressure, and pressurizer level were normal
- Plant was stable with Integrated Control System (ICS) in automatic
- Power was 95% by computer heat balance
- Nuclear Instrument (NI) power indicated 98%
- Flux Imbalance was -2.0%

The flux/delta flux/flow trip uses flux imbalance signals from nuclear instrumentation and Reactor Coolant System (RCS) flow signals from primary instrumentation to modulate a flux level trip setpoint. The result is a variable trip setpoint based on instantaneous values of flux imbalance or flow. The technical specification setpoint limit is 106.8% power based on flow of 100% with a flux imbalance range of -18.2% to +18.2%.

The licensee conducted a post trip analysis to determine the cause of the reactor trip and provide corrective action if required. The following was identified:

- The trip setpoint voltages in the Reactor Protection System (RPS) were lower than required. The voltages measured corresponded to trip setpoints of 102% power at 100% flow.

- RCS flow was 98% due to the use of BPRAs in the reload core. The use of the BPRAs reduced core bypass flow, raised the differential pressure across the core, and reduced loop flow. Since loop flow was reduced, the voltage input to the RPS was reduced and a corresponding reduction in trip setpoint automatically occurred.

- New flow transmitters were installed during the refueling outage and sensitivity and noise were creating an output that oscillated by as much as two percent. This oscillation in flow signal caused a corresponding oscillation in the trip setpoint.

The licensee determined that the low 100% flow setpoint in existence, the reduction in setpoint due to 98% flow, and the further periodic reduction caused by flow signal oscillations were cumulative. On a flow signal negative oscillation, the existing 98% flux signal exceeded the trip setpoint and the reactor tripped.

The licensee has taken the following corrective action:

- Reduced plant power to 90% by heat balance to avoid further challenges to the protection system.

- Conducting an investigation into an event at the Rancho Seco station in which all four NI channels simultaneously spiked high resulting in a high flux trip. This event is somewhat similar in that all four protection channels tripped within milliseconds of each other.

- Contacting the vendor (Rosemount) for assistance in reducing the noise and oscillations in the transmitter output.

- A technical specification change has been requested which will allow credit for the increased core flow and use of the present 98% indicated flow as the 100% flow baseline. The bistable trip setpoint will then be readjusted.

The inspector reviewed the following items:

IC 2000.06, "Setting of the RPS Flux/Delta Flux/Flow Trip Bistable Setpoints"

ST 5030.18.06, "Check of RPS Flux/Delta Flux/Flow Bistable Setpoint"

Post Trip Analysis Checklist

The inspector completed independent setpoint calculations and verified that a flow oscillation to 96% indicated flow would result in a trip setpoint less than 98% indicated flux level. Since contributing factors (low full-flow trip setpoint, automatic setpoint reduction due to low indicated flow, and nuclear instruments indicating higher than heat balance power) were conservative the inspector considers that this event

posed no threat to the health and safety of the public.

No items of noncompliance or deviations were identified.

8. Core Power Distribution

Below is a list of procedures/test packages for which the inspectors have completed their review:

PT 5175.02.03, "Core Power Distribution"

ST 5020.01.06, "Imbalance, Tilt, and Rod Index - Group 38 Alarms Inoperable"

ST 5021.01.03, "Determination of Hot Channel Factors"

ST 5030.10.04, "RPS Monthly Imbalance Check"

Note: PT 5175.02.03 was performed at 40%, 75% and 93% power.

These procedures/test packages were reviewed against the Final Safety Analysis Report (FSAR), Safety Evaluation Report (SER), applicable Regulatory Guides and Standards, and portions of 10 CFR 50. The inspectors had the following comments:

a. The limit for the nuclear enthalpy rise hot channel factor for PT 5175.02.03 at 75% power was incorrectly calculated.

   This was subsequently recalculated by the licensee and determined to have no impact on the acceptability of the test. This simple error indicates a lack of effective review and is considered to be a violation of 10 CFR 50, Appendix B, Criterion XI, and an example of an item of noncompliance (346/85009-01d(DRS)).

b. Deviation for radial peaking and total peaking factors is determined by comparing the highest measured value against the highest predicted value over an eighth-core map. This frequently results in comparing the measured values of one fuel assembly to the predicted values of a different assembly.

   The inspector questioned the purpose and validity of this evaluation. The licensee stated that this was the method used by the reactor vendor, Babcock and Wilcox, and referred the question to them. Receipt and evaluation of the Babcock and Wilcox response by the inspector and concurrence by the Office of Nuclear Reactor Regulation is an open item (346/85009-05(DRS)).

No additional items of noncompliance or deviations were identified.

9. Incore-Excore Calibration

The inspectors reviewed the test package for the following periodic test:

PT 5175.01, "Power Imbalance Detector Correlation"

The test package was reviewed against the FSAR, SER, applicable Regulatory Guides and Standards, and portions of 10 CFR 50.

No items of noncompliance or deviations were identified.

10. Estimated Critical Boron Concentration Calculations

a. On April 13, 1985, during a reactor startup, the reactor went critical at a regulating rod position of approximately 180% withdrawn (300% withdrawn is equivalent to all three regulating rod groups being fully withdrawn) which was before it was predicted to go critical based upon a computer printout dated April 8, 1985. The printout indicated an estimated critical regulating rod position of 192 to 273% WD (withdrawn). As required, the operators reinserted the regulating rods until they were fully inserted into the core and contacted the Nuclear and Performance Engineer (NPE). Based upon the fact that there was a difference of approximately 40 ppm between predicted and actual all rods out boron concentration, as documented in Surveillance Test ST 5010.03 for Cycle 5, the NPE recommended that the reactor coolant system boron concentration be increased by approximately 40 ppm. After making this adjustment, the reactor went critical within the limits of the estimated critical regulating rod position.

   Technical Specification 4.1.1.1.2 required that the overall core reactivity balance be compared to predicted values to demonstrate agreement within ±1% $\Delta K/K$ at least once per 31 Effective Full Power Days (EFPD). The inspector reviewed the daily reactivity anomalies log for Cycle 5 (from January 23, 1985 to April 23, 1985) and noted that the largest reactivity error during steady state power operation was 0.496% $\Delta K/K$; consequently the Technical Specification requirement was met. Technical Specification 4.1.1.1.2 also requires that the predicted reactivity values be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPDs after each fuel loading. Since the fuel burnup on April 15, 1985, was only about 50 EFPDs, this requirement did not apply during the April 13, 1985, reactor startup.

b. On April 25, 1985, during a reactor startup, the reactor operators pulled the regulating rods to the maximum allowable estimated rod position and the reactor did not go critical. The operators contacted the Nuclear and Performance Engineer and, upon reviewing the computer printout of the estimated critical rod position calculation, the NPE discovered that an input value for power history was incorrect. Based upon the incorrect value it appeared

that the reactor had tripped at midnight on April 24, 1985, whereas it had actually tripped at 4:00 a.m. The NPE corrected the power history input and reperformed the estimated critical rod position calculation. In the reactor startup that followed, the reactor went critical within the range specified by the corrected calculation.

While the causes of the above events were unrelated, the inspector cautioned the licensee that greater care related to calculation of estimated critical rod position may be needed. The licensee's performance in this area will be examined in subsequent NRC inspections.

No items of noncompliance or deviations were identified.

11. Open Items

Open items are matters which have been discussed with licensee which will be reviewed further by the inspectors, and which involve some action on the part of the NRC or licensee or both. Open items disclosed during the inspection are discussed in Paragraphs 2.b.(2), and 8.b..

12. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. An unresolved item disclosed during the inspection is discussed in Paragraph 6c.

13. Exit Interviews

The inspectors met with the licensee representatives denoted in Paragraph 1 on April 26, 1985, and at the conclusion of the inspection by telephone on May 1, 1985. The inspectors summarized the scope of the inspection and the findings. The licensee acknowledged the statements made by the inspectors with respect to open and unresolved items and the items of noncompliance denoted in Paragraphs 2.a., 2.b.(1), 5.a., 6.b. and 8.a.. The inspectors also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspector during the inspection. The licensee did not identify any such documents or processes as proprietary.