U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report Nos. 50-315/93013(DRP); 50-316/93013(DRP)

Docket Nos. 50-315; 50-316 License Nos. DPR-58; DPR-74

Licensee: Indiana Michigan Power Company
1 Riverside Plaza
Columbus, OH 43216

Facility Name: Donald C. Cook Nuclear Power Plant, Units 1 and 2

Inspection At: Donald C. Cook Site, Bridgeman, MI

Inspection Conducted: May 1, 1993 through June 15, 1993.

Inspectors: J. A. Isom
D. J. Hartland
R. C. Paul
E. R. Schweibinz

B.C. McCabe

Approved By: B. C. McCabe, Acting Chief
Reactor Projects Section 2A

7/1/93
Date

Inspection Summary: Inspection from May 1, 1993, through June 15, 1993. (Report Nos. 50-315/93013(DRP); 50-316/93013(DRP))

Areas Inspected: Routine, unannounced inspection by the resident and region-based inspectors of plant operations; maintenance and surveillance; radiological controls; reportable events; balance of plant (BOP) and reactor protection system (RPS) instrument replacement; and corrective actions on previously identified items. In addition, a routine management meeting was held at the NRC Region III office on May 13, 1993.

Results : No violations or deviations were identified in any of the six areas inspected.

The inspection disclosed a strength in the licensee's conduct of the maintenance activity involving the FRV-240 hand/auto controller.

The inspection disclosed weaknesses in the conduct of maintenance activities involving the repair of valve 1-QRV-171 and the setpoint change to the T Avg deviation alarm.

DETAILS

1. Persons Contacted

a. Management Meeting - May 13, 1993

American Electric Power Service Company(AEPSC)

E. Fitzpatrick, Vice President
J. Kobyra, Manager, Electrical Systems Division
W. Smith, Jr., Chief Nuclear Engineer
S. Brewer, Manager, Nuclear Safety & Licensing
P. Barrett, Director, Quality Assurance
D. Williams, Manager, Radiological Support
M. Ackerman, Nuclear Licensing Engineer
B. Signet, Senior Attorney
D. Malin, Manager, Nuclear Licensing
G. Lewis, Group Manager, Operations Support
W. Sotos, Senior Engineer
R. Simms, Section Manager, Assessment

Indiana Michigan Power/Cook Nuclear Power Plant

A. Blind, Plant Manager
M. Barfeltz, Nuclear Safety Analysis Supervisor
J. Wiebe, Superintendent, Safety & Assessment
R. Gillespie, Plant Scheduling Superintendent

U.S. Nuclear Regulatory Commission, Region III

E. G. Greenman, Director, Reactor Projects
T. O. Martin, Acting Director, Division Reactor Safety
W. D. Shafer, Chief, Reactor Projects Branch 2
W. M. Dean, Acting Director, Project Directorate III-1, NRR
J. A. Isom, Senior Resident Inspector, D. C. Cook
E. R. Schweibinz, Senior Project Engineer
G. M. Hausman, Reactor Inspector
W. D. Pegg, Reactor Inspector
S. S. Lee, Project Engineer, NRR

b. Inspection - May 1 through June 15, 1993

*A. A. Blind, Plant Manager
*K. R. Baker, Assistant Plant Manager-Production
L. S. Gibson, Assistant Plant Manager-Projects
*J. E. Rutkowski, Assistant Plant Manager-Technical Support
B. A. Svensson, Executive Staff Assistant
*T. P. Beilman, Maintenance Superintendent
P. F. Carteaux, Training Superintendent
D. L. Noble, Radiation Protection Superintendent
L. J. Matthias, Administrative Superintendent

T. K. Postlewait, Design Changes Superintendent
S. A. Richardson, Operations Superintendent
P. G. Schoepf, Project Engineering Superintendent
*J. S. Wiebe, Safety & Assessment Superintendent
L. H. Vanginhoven, Site Design Superintendent
G. A. Weber, Plant Engineering Superintendent
D. C. Loope,Chemistry Superintendent
*M. L. Horvath, Quality Assurance Supervisor

The inspector also contacted a number of other licensee and contract employees and informally interviewed operations, maintenance, and technical personnel.

*Denotes some of the personnel attending the Management Interview on June 11, 1993.

2. Plant Operations (71707, 71710, 42700)

The inspector observed routine facility operating activities conducted in the plant and the main control rooms. The inspector monitored the performance of licensed Reactor Operators, Senior Reactor Operators, Shift Technical Advisors, and Auxiliary Equipment Operators (AEOs) including procedure use and adherence, records and logs, communications, and the degree of professionalism of control room activities.

The inspector reviewed the licensee's evaluation of corrective action and response to off-normal conditions. This included compliance with reporting requirements.

The inspector noted the following with regard to the operation of Units 1 and 2 during this reporting period:

a. Unit 1 Status

The licensee operated the unit at full power throughout the inspection period, with no significant operational problems noted.

b. Unit 2 Status

The licensee operated the unit at full power throughout the inspection period until June 11, 1993, when reactor power was reduced to 70 percent power. The licensee intends to operate the unit at that power level for the remainder of the cycle in order to separate the two units' scheduled 1994 refueling outages.

c. ENS Notification

On June 9, 1993, the licensee made a one hour ENS notification after they discovered a condition that was outside the design basis of the plant. The condition was related to fire doors to the turbine driven auxiliary feedwater pump rooms and to the adjacent hallway. These doors were left open to provide a vent

path for steam that would be released in the event of a rupture of the steam lines that supply the turbine driven pumps. The licensee discovered, during review of their EQ evaluation for the auxiliary feedwater (AFW) system, that the steam from the rupture could melt the fusible links on these doors, causing the doors to close. The closed doors would eliminate the vent path, and pressure in the rooms would increase. As short-term corrective action, the licensee blocked the doors open and initiated compensatory fire watches. The inspector will review the licensee's root cause evaluation and corrective action to be documented in the LER to be submitted to the NRC in response to the event.

d. Simulator/Procedure Evaluation (71707, 42700)

During the period May 10-12, 1993, a group of NRC inspectors including the NRC Region III Projects Branch Chief and the Senior Project Engineer, joined the D. C. Cook Resident and Senior Resident Inspectors for an evaluation of selected procedures on the plant control room simulator.

Selected procedures and simulator operations were used to familiarize the NRC team with plant response during certain evolutions. Additionally, the adequacy of the procedures applying to the circumstances were assessed. Licensee support consisted of two professional staff trainers and free access to the simulator during the three day evaluation. The trainers operated the simulator scenarios and provided guidance on hardware and procedures.

The following procedures were covered by the evaluation:

- "Reactor Startup," *2-OHP-4021.001.002, Revision 15, February 26, 1993.

- "Power Escalation," *2-OHP-4021.001.006, Revision 11, March 29, 1993.

- "Turbine Generator Normal Startup and Operation," *2-OHP 4021.050.001, Revision 5, October 25, 1990.

- "Reactor Trip or Safety Injection," 02-OHP-4023.E-0, Revision 6, December 16, 1992.

- "Faulted Steam Generator Isolation," 2-OHP 4023.E-2, Revision 1, May 19, 1989.

- "Loss of Reactor or Secondary Coolant," 02-OHP- 4023.E-1, Revision 6, December 30, 1992.

- "SI Termination," 02-OHP-4023.ES 1-1, Revision 4, December 16, 1992.

- "Steam Generator Tube Rupture," 02-OHP-4023.E-3, Revision 4, December 30, 1992.

- "Loss of RHR While Partially Drained or at Mid Loop," 02-OHP-4022.017.001, Revision 1, December 22, 1992.

No substantive discrepancies were identified in any of the procedures. Some comments were provided to the licensee representatives for their consideration and, if deemed appropriate, further followup action.

No violations, deviations, unresolved or inspector followup items were identified.

3. Maintenance/Surveillance (62703, 61726, 42700)

The inspector reviewed maintenance activities as detailed below. The focus of the inspection was to assure the maintenance activities were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with Technical Specifications. The following items were considered during this review: the Limiting Conditions for Operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures; and post maintenance testing was performed as applicable.

The following activities were inspected:

a. T Avg Deviation Alarm Setpoint Change

The inspector reviewed activities associated with a setpoint change to the reactor coolant loop auctioneered average temperature deviation bistables on Unit 2. The inspector noted that the initial attempt to perform the setpoint change resulted in a plant transient, which was caused by inadequate work order review and a lack of questioning attitude of the personnel involved as described below.

The licensee changed the setpoint to alarm at 4 degrees instead of 3 degrees F to eliminate a standing alarm in the control room. The alarm had come in due to a higher than expected loop 3 T Avg, the cause of which the licensee was still investigating. The setpoint change was non-safety related, as the bistables provided only an alarm function and were not part of the reactor protection circuitry.

The inspector observed the licensee's second attempt at performing the setpoint change. During the initial attempt on May 13, 1993, I&C used "T Avg/Auctioneered T Avg Deviation Alarms Calibrated," 2IHP6030IMP.203, Revision 6, dated May 13, 1993, which, when used to calibrate all four loops, was not intended to be performed during power operations. The procedure required that the leads be

lifted on all four loops at the same time. When I&C did this, all T Avg channels failed low, which ultimately resulted in the automatic tripping of the pressurizer heaters and a letdown isolation. I&C immediately terminated use of the procedure and restored the T Avg channels. The licensee also put an administrative hold on the calibration procedure.

The inspector reviewed the calibration procedure and noted that it did not contain any precautions to prevent the removal of all four loops at one time at power. In addition, operations and I&C personnel involved in performing the activity did not question the ramifications of removing all four loops. Also, the inspector noted during review of a work order dated back in 1991 that the licensee had identified the procedural deficiency and did not take any corrective action at that time. Due to the minimal safety significance involved, the inspector determined that the licensee's failure to take action was not a violation of regulatory requirements.

During the second attempt performed on May 18, 1993, I&C used "Generic Calibration Procedure," 12IHP6030.IMP.066, Revision 1, dated July 23, 1992, to calibrate the bistables one at a time. As this procedure provided only general guidelines, I&C used primarily "skill of the trade" to perform the setpoint change. I&C successfully completed the calibration at that time, and the inspector did not observe any deficiencies.

b. Turbine Driven Auxiliary Feedwater Pump (TDAFP) Bearing Inspection

The inspector observed the disassembly and inspection of the TDAFP outboard thrust bearing and adjustment to the balance drum setting. The inspector noted that the maintenance personnel involved in the activity were knowledgeable and that the work observed was performed satisfactorily.

The licensee initiated the inspection after an increasing trend in bearing oil temperature was observed, and as part of their investigation into the motor driven auxiliary feedwater pump bearing failure discussed in IR 50-315/93011(DRP); 50-316/93011(DRP). Upon disassembly, maintenance did not identify any apparent damage to the bearings. However, they did observe during removal of the bearing from the shaft that it slid off easier than what had been experienced in the past.

In response, maintenance measured the OD of the pump shaft and the ID of the bearing and, after consultation with the vendor, determined that the shaft was undersized by .0016". Although the vendor endorsed continued use of the shaft, they recommended that it be repaired "at the nearest opportunity." Since there was no indication that the shaft had been turning within the bearing

inner race, as evidenced by the absence of any apparent damage, the licensee intended to monitor bearing temperature and vibration during future pump runs.

The licensee attributed the elevated bearing oil temperature to an excessive gap between the balance drum and balance sleeve that was discovered during the inspection, rather than to the undersized shaft. The balance drum functioned to counter-act axial force initiated by fluid flow through the pump. The excessive gap resulted in transfer of this load to the thrust bearing. The as-found drum setting was .008". Maintenance adjusted the gap to within the required .002 to .005" tolerance limit. The licensee was still investigating why the as-found gap setting was out of tolerance. The inspector will continue to monitor the licensee's investigation into the AFW pump bearing issue as followup to unresolved item # 50-315/93011-03; 50-316/93011-03.

c. Feedwater Regulating Valve (FRV) Controller Replacement

The inspector observed the replacement of the hand/auto controller for FRV 240 and reviewed the associated work procedure "Manual Handwheel Operation of 1-FRV-240," 1OHP SP.103, Revision 0, dated May 13, 1993. The inspector noted that, overall, the procedure was well-written and that the evolution was well-planned and completed successfully without incident.

The licensee entered TS 3.0.3 during the controller changeout because the automatic close safety feature of the FRV from a feedwater isolation signal was inhibited. The licensee transferred control of the valve to local handwheel operation during the evolution. The licensee completed the evolution within one hour; therefore, a TS-required unit shutdown was not initiated.

During review of the procedure prior to the evolution, the inspector noted that instructions to replace the controller were contained in a job order which was referenced in the procedure. The procedure included only the instructions for transferring control of the valve between the controller and the handwheel. The inspector identified a concern that since the instructions for controller change-out were not in the procedure, they would not get same level of review as the procedure. The inspector communicated the concern to the licensee, who subsequently added the instructions to the procedure as an attachment.

The procedure required that continuous communications be established between the control room operator at the steam generator water level control station and the operator performing manual manipulation of the FRV locally. The inspector observed, however, that direct communication was maintained between the two

operators only when valve control was being transferred. This did not result in any problems, however, and coordination between the control room and at the valve location was good.

The inspector also observed that the licensee had installed a micrometer on the valve stem in order to detect stem movement. The System Engineer also instructed the local valve operator to hang onto the valve handwheel during the evolution to prevent any inadvertent handwheel movement due to vibration. These instructions were not contained in the procedure; however, the licensee intends to add them to the next revision.

d. QRV-171 Repair

The inspector observed activities related to the repair of 1-QRV-171, excess letdown heat exchanger outlet valve. The inspector observed that the licensee experienced several problems during the activity due primarily to poor job planning which resulted in delays and higher than expected man-rem exposure.

1-QRV-171 was an air-operated, three-way valve which directed flow from the excess letdown heat exchanger to either the reactor coolant drain tank (RCDT) or the volume control tank (VCT). The valve was normally positioned to the VCT and designed to fail in that position; however, during an evolution several months ago in which excess letdown was used to add water to the RCDT, the licensee was unable to stroke the valve back to the fully-seated VCT position. Therefore, in the event that excess letdown was needed, some flow would be diverted to the RCDT, which would represent unidentified reactor coolant system (RCS) leakage.

In order to initiate the repair, the licensee secured RCS letdown in order to reduce radiation levels in the regenerative heat exchanger room, where the valve was located. General radiation levels inside the room with letdown secured were approximately 300 mr/hr. In addition, the licensee diverted reactor coolant pump (RCP) seal leakoff return flow to floor drains to isolate QRV-171.

Among the problems encountered was communication between the control room and the containment. Although a procedure prerequisite required that communication equipment be made available, the inspector observed that at least one of the headphone jacks inside containment did not work, which resulted in loss of direct communication when the System Engineer was in the area which required use of the jack.

In addition, the licensee did not have a contingency plan regarding availability of replacements for the primary individuals involved, in the event that the duration of the job went beyond the scheduled 6 hours. The total evolution lasted approximately 15 hours, and the inspector observed that some AEO's had to make multiple containment entries due to lack of additional support.

In addition, the System Engineer, who was coordinating the activity, exited containment about 5 hours into the evolution to take a short break, but was delayed in returning for several hours after he surveyed positive for radioactive contamination. As a result, there was not a System Engineer present to coordinate activities inside containment for several hours.

To divert the RCP seal leak-off return flow, the licensee installed a manifold on a vent line on each return line. The licensee diverted flow by opening a temporary valve on each manifold while simultaneously closing the seal return isolation valve. The inspector noted that there was good coordination between the valve operators and the control room during the valve manipulation to control seal water back pressure/flow. However, spikes in the RCP #2 return flow were indicated on the control room instrumentation on several occasions for unknown reasons. This caused some distraction in the control room. In addition, the hose connected to the RCP #2 leak-off came loose twice during the evolution, spraying radioactive water and providing unnecessary distractions. The inspector noted that there was no radiation protection (RP) support in containment at the time of the second occurrence to ensure that appropriate contamination control practices were implemented. During followup to this concern, the inspector was told by the licensee that there were no activities in progress inside containment that warranted RP support at the time of the second concurrence.

The inspector also observed that there was a delay in locating 1-CA-538, control air isolation valve to 1-QRV-171. The licensee also experienced delays due to tool/component unavailability.

Overall, the inspector observed that the quality of the maintenance work on the valve was good. Upon disassembly and removal of the valve internals, the licensee discovered a small screw inside the valve body which prevented the valve from stroking to the VCT position. After removing the screw and reassembling the valve, the licensee stroked the valve and discovered that it would still not seat in the VCT position. The System Engineer observed the valve movement and determined that a simple adjustment to the stroke was necessary. However, the procedure did not provide contingent instructions to make the adjustment; therefore, the licensee returned the valve to the RCDT position and suspended use of the procedure. On May 28, 1993, the licensee set the stroke on the valve and successfully seated it in the VCT position.

No violations, deviations, unresolved or inspector followup items were identified.

4. Radiological Controls (71707)

During routine tours of radiologically controlled plant facilities or areas, the inspector observed occupational radiation safety practices by the radiation protection staff and other workers.

Effluent releases were routinely checked, including examination of on-line recorder traces and proper operation of automatic monitoring equipment.

Independent surveys were performed in various radiologically controlled areas.

a. Radioactive Waste Storage

The inspector toured the site to inspect contaminated material and waste storage facilities; independent surveys were performed during the inspection. Within the controlled area are two 16,000 gallon tanks with low level contaminants and glycol awaiting the results of an approved disposal methodology, several sea vans containing equipment awaiting shipment to a contract vendor for segregation, compaction, and decontamination, a trash dumpster containing several hundred gallons of low level contaminated turbine building sludge, and several waste oil storage tanks with low level contamination awaiting disposal. Areas inspected outside the radiologically controlled area were the Contaminated Equipment Storage Area (CESA), and the Radioactive Material building (RAM). All facilities and storage areas were properly posted and controlled.

CESA

The CESA building stores contaminated equipment and tools; waste is not stored in the building. The fence surrounding the CESA area and the building are maintained locked and only authorized personnel have keys; a radiation work permit is required for access. Sea vans and storage containers also used for equipment storage are located outside the building, but within the locked fence area. Through a review of procedures, tours of the CESA and surrounding area, and discussions with personnel, it appears the facility is controlled and operated in accordance with requirements. The inspector also reviewed the licensee's evaluation of the facility in which they concluded that use of the CESA did not constitute an unreviewed safety question as defined in 10 CFR 50.59; no problems were identified.

RAM

The RAM is a concrete building with a heating, ventilation, and air conditioning system, radiation monitors, and fire protection equipment. The building is used as the interim radwaste storage facility for resins, dry active waste, and metal oxides. Based on

the inspection, it appeared the operation and control of the facility is in accordance with the requirements and consistent with the safety analysis description.

No violations, deviations, unresolved, or inspector followup items were identified.

5. Reportable Events (92700, 92720)

The inspector reviewed the following Licensee Event Reports (LERs) by means of direct observation, discussions with licensee personnel, and review of records. The review addressed compliance with reporting requirements and, as applicable, that immediate corrective action and appropriate action to prevent recurrence had been accomplished.

a. (Closed) LER 315/92007-LL: Containment Type "B" and "C" Leakage Exceeds Limiting Condition for Operation (LCO) Value

This LER was closed based on adequate licensee root cause evaluation and corrective action.

The licensee issued the LER to document exceeding the Technical Specification (TS) LCO value for Type "B" and "C" leakrate tests on containment penetrations during the 1992 Unit 1 refueling outage. The licensee repaired the affected valves and successfully completed as-left testing. The inspector reviewed the licensee's root cause evaluation and corrective action for each of the valve failures and found them to be adequate. The licensee determined that none of the failures would have resulted in any additional leakage from containment.

In addition, the licensee performed maintenance activities on two valves prior to obtaining as-found leak-rates. The testing was originally scheduled to be performed prior to the maintenance activity; however, the testing was subsequently rescheduled during the outage and the two activities were not coordinated properly. As corrective action, the licensee was in the process of revising their scheduling software interface to enhance identification of proceeding and succeeding requirements for work activities.

b. (Closed) LER 316/92002-LL: Containment Isolation Valves Not Repaired When ASME Section XI Leak Rate Acceptance Criteria Exceeded

This LER was closed based on adequate licensee root cause evaluation and corrective action.

In January 1992, during review of in-service testing (IST) valve program test data, the licensee discovered that two Unit 2 boron injection tank outlet valves, 2-ICM-250 and 2-ICM-251, had been returned to service following the 1989 Unit 2 SG replacement outage with seat leakage in excess of ASME Section XI limits. The

valves remained in service until August 1990, when the unit was shutdown for refueling. The condition was a condition prohibited by TS 4.0.5., which specified IST surveillance requirements. However, the licensee determined that the valves were capable of performing their intended safety function.

The licensee determined the root cause to be a deficiency in their containment local leak (LLRT) test procedure. Although the licensee tested some Section XI valves per the LLRT procedure, the procedure included only LLRT acceptance criteria which, in some instances, were less restrictive than Section XI limits. In addition, the licensee did not perform an independent review of test data for Section XI purposes.

As corrective action, the licensee revised the affected procedures to include Section XI limits. Additionally, the IST Program Engineer now performs an independent review of LLRT data for those valves in the IST valve program.

This event involved a violation of regulatory requirements. However, it was identified by the licensee and was properly reported and corrected. In addition, the event had minimal safety significance. Therefore, pursuant to the NRC enforcement policy (10 CFR 2, Appendix C), the NRC is exercising enforcement discretion for this matter, and no Notice of Violation will be issued.

No violations, deviations, unresolved, or inspector followup items were identified. One non-cited violation was identified.

6. <u>Balance of Plant (BOP) and Reactor Protection System (RPS) Instrument Replacement (37700, 71702)</u>

D. C. Cook decided to replace their RPS with a "state of the art" solid state system from Taylor (later bought out by Combustion Engineering (CE)). Because of some incompatibilities with the systems, the licensee was forced to delay the replacement. The RPS contract was switched to Foxboro. Taylor (CE) will supply the BOP replacements. Inspection of the BOP replacement activities was undertaken to observe the standard of quality applied to these systems and to use this information as an input to the inspection resource allocation process for the RPS replacement.

a. Selected BOP systems were reviewed to see how controls were placed on the modifications to ensure that the system with instruments being replaced would remain functionally the same. Elementary wiring diagrams for several instrument loops were selected for a more indepth review. Factory acceptance test procedures were reviewed along with observation of wiring and calibration testing in the temporary staging area being used for this purpose. The inspector found the above controls and activities acceptable and performed in a professional manner.

Administrative control of the configurator (used to upload and download setpoints into the digital controllers) was reviewed. The controls included both hardware and software protection and were found to be acceptable. The inspector observed several download and upload operations, and the methods used to verify that the proper setpoints were being placed in the correct units. Several random checks were made by the inspector of the data (setpoints) provided from the Engineering office (Columbus, OH) to the data that was placed in the controller. Line-for-line checks were made and no discrepancies were noted.

Point to point wiring checks were observed while the equipment was energized and prior to calibration checks. In one case, the inspector and the testing engineers observed smoke from behind a panel. It appeared to be coming from a 10 ohm resistor in parallel with a test point. This test point (FP-121) was part of the Unit 2 pressurizer level control system in Cabinet GG18, shown on drawing 2985542-1. A controlled system for logging any such errors or discrepancies was used. Very few such errors had been found which indicated that the factory testing was thorough. Calibration checks were also observed to be done with controlled procedures and with calibrated test equipment.

b. The replacement of the Reactor Protection System (RPS) instrumentation is with Foxboro, SPEC 200 and SPEC 200 MICRO equipment. An audit of the factory acceptance test was performed at the Foxboro Company factory in Massachusetts on May 4-7, 1993. The audit team consisted of a team leader from the Instrumentation and Controls Branch, a Senior Project Engineer from Region III, and three contractors from the Idaho National Engineering Laboratory (INEL). The team leader from NRR will document the results of this audit in separate correspondence.

Areas reviewed by the inspector included the development of pseudo code and review of the documentation of the structured walkthroughs for four software blocks that will be used at D. C. Cook. The block identification and date of the walkthroughs are CALC (April 1, 1985), GATE (April 24, 1985), LLAG (May 9,1985), and ALRM (June 13, 1985). Also, a followup independent review was conducted on March 24, 1989. The inspector verified that in all cases but one, the action items identified in the walkthroughs had been subsequently incorporated into the Code, Functional Specification (Corporate Product Specification), Program Technical Description, or other appropriate documents. The March 24, 1989, followup independent review correctly identified the only case in which an action item (Action Item 2, identified in the ALRM walkthrough), should not have been incorporated, and the inspector verified the appropriateness of this decision.

On May 7, 1993, the inspector observed factory acceptance testing of equipment that will be used for Unit 2 RPS replacement. The testing was performed in a controlled manner using approved

procedures with calibrated test equipment. No discrepancies were noted in any of the activities observed.

No violations, deviations, unresolved, or inspector followup items were identified.

7. Actions on Previously Identified Items (92701, 92702)

(Closed) Unresolved Item (50-315/316-93008-02(DRS)) This item concerned the use of pump flow curves for the centrifugal charging (CC) pumps in lieu of a fixed reference value as required by ASME Code Section XI. The licensee responded to this item by letter from E. E. Fitzpatrick, dated May 21, 1993. The test procedure acceptance criteria in the Technical Data Book was revised to a fixed reference value of 125 gpm for the CC pumps. The inspector noted a tolerance band, however, it was also included on the reference value to allow flow to be adjusted between 120-130 gpm. As stated in inspection follow-up item (IFI) (50-315/316-93008-03(DRS)), tolerance bands were not allowed by the Code, although guidance being developed by NRR on this issue was discussed with the licensee. This guidance would allow up to a 2 percent band on the reference value in some cases without relief from the Code. The 4 percent band established for the CC pumps exceeded this limit. The licensee will review the use of tolerance bands on the fixed reference value under the above mentioned IFI. This item is considered closed based on establishing a fixed reference value.

8. Management Meeting

A management meeting, attended as indicated in paragraph 1.a., was conducted in the NRC Region III office on May 13, 1993. The purpose of the meeting was to discuss information relative to the licensee's graded root cause program, integrated work management and nuclear plant maintenance program, self assessments, and RPS and BOP control process instrument replacement.

9. Management Interview

The inspectors met with licensee representatives (denoted in paragraph 1.b.) on June 11, 1993, to discuss the scope and findings of the inspection. In addition, the inspector 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. However, the licensee did not agree with the inspector's assessment as to significance of the System Engineer's absence from containment during the QRV-171 evolution, as discussed in paragraph 3.d of this report.