U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-315/90006(DRP); 50-316/90006(DRP)

Docket Nos. 50-315; 50-316 Licenses No. 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, Bridgman, Michigan

Inspection Conducted: February 7 through March 20, 1990

Inspectors: B. L. Jorgensen

F. A. Maura

D. G. Passehl

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Approved By: Bruce L. Burgess, Chief
Reactor Projects Section 2A

3/29/90
Date

Inspection Summary

Inspection on February 7 through March 20, 1990 (Reports No. 50-315/90006(DRP); No. 50-316/90006(DRP))

Areas Inspected: Routine unannounced inspection by resident and Region III inspectors of: plant operations; maintenance; surveillance; engineering and technical support; safety assessment/quality verification; emergency preparedness; reportable events; Bulletins, Notices and Generic Letters; and, NRC Region III requests. Also, a management meeting was conducted at the site on March 15, 1990, primarily to discuss maintenance issues. The following Safety Issues Management System (SIMS) items were reviewed, with the indicated results: (Closed) NRC Bulletins 88-02 through 88-09 inclusive.

Results: Of the nine areas inspected, no violations or deviations were identified in eight areas. Two violations which the licensee had identified (failure to perform required fire protection compensatory measures - Paragraph 8) were noted in the remaining area. Both were caused by miscommunications, which in turn led to misjudgments, concerning administrative controls for fire protection equipment. Pursuant to 10 CFR 2, Appendix C, Section V.A. no Notice of Violation is being issued concerning these events, as discussed at Paragraph 8.

The inspection disclosed no new, notable strengths or weaknesses in the areas inspected. One new Unresolved Item was identified (and is discussed in Paragraph 5.b) in the area of engineering/technical justification for systems classifications under the containment leak-testing program.

DETAILS

1. Persons Contacted

   a. Management Meeting - March 15, 1990

      (1) Nuclear Regulatory Commission

      A. B. Davis, Regional Administrator, NRC Region III
      J. A. Zwolinski, Assistant Director For Region III and V Reactors, NRR
      H. J. Miller, Director, Division of Reactor Safety, Region III
      B. Burgess, Chief, Projects Section 2A, Region III
      J. G. Giitter, Licensing Project Manager, NRR
      J. Wechselberger, Senior Operations Engineer, NRR
      M. Dapas, Operations Engineer, NRR
      E. Schweibinz, Project Engineer, Section 2A, Region III
      B. Jorgensen, Senior Resident Inspector, Region III
      D. Passehl, Resident Inspector, Region III

      (2) American Electric Power/Indiana Michigan Power

      D. H. Williams, Jr., Senior Executive Vice President, AEP
      M. P. Alexich, Vice President, Nuclear Operations, AEPSC
      T. Argenta, Assistant Vice President, Nuclear Engineering, AEPSC
      P. Barrett, Director of Quality Assurance, AEPSC
      S. Brewer, Manager, Nuclear Safety and Licensing, AEPSC
      J. Kurgan, Section Manager, Nuclear Operations, AEPSC
      S. Klementowicz, Section Manager, Radiological Support AEPSC
      A. Blind, Plant Manager
      J. Rutkowski, Assistant Plant Manager, Technical Support
      K. Baker, Assistant Plant Manager, Production
      B. Svensson, Executive Staff Assistant
      J. Sampson, Operations Superintendent
      T. Bielman, Maintenance Superintendent
      J. Droste, Technical Superintendent, Engineering

   b. Inspection - February 7 through March 20, 1990

   A. Blind, Plant Manager
   J. Rutkowski, Assistant Plant Manager, Technical Support
   L. Gibson, Assistant Plant Manager, Projects
   *K. Baker, Assistant Plant Manager, Production
   B. Svensson, Executive Staff Assistant
   J. Sampson, Operations Superintendent
   E. Morse, QC/NDE General Supervisor
   T. Beilman, Maintenance Superintendent
   J. Droste, Technical Superintendent, Engineering
   T. Postlewait, Design Changes, Superintendent
   *P. Carteaux, Safety and Assessment Superintendent
   J. Wojcik, Technical Superintendent, Physical Sciences

M. Horvath, Quality Assurance Supervisor
D. Loope, Radiation Protection Supervisor
*H. Runser, Operations Production Supervisor
*D. Fitzgerald, Environmental Supervisor
*M. Mierau, STA Supervisor
*M. Barfelz, Senior Performance Engineer

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 March 21, 1990.

2. <u>Operational Safety Verification (71707, 71710, 42700)</u>

Routine facility operating activities were observed as conducted in the plant and from the main control rooms. Plant startup, steady power operation, plant shutdown, and system(s) lineup and operation were observed as applicable.

The performance of licensed Reactor Operators and Senior Reactor Operators, of Shift Technical Advisors, and of auxiliary equipment operators was observed and evaluated including procedure use and adherence, records and logs, communications, shift/duty turnover, and the degree of professionalism of control room activities. The Plant Manager, Assistant Plant Manager-Production, and the Operations Superintendent were well-informed on the overall status of the plant, made frequent visits to the control rooms, and regularly toured the plant.

Evaluation, corrective action, and response to off-normal conditions or events, if any, were examined. This included compliance with any reporting requirements.

Observations of the control room monitors, indicators, and recorders were made to verify the operability of emergency systems, radiation monitoring systems and nuclear reactor protection systems, as applicable. Reviews of surveillance, equipment condition, and tagout logs were conducted. Proper return to service of selected components was verified.

a. Unit 1 operated routinely at design power levels until March 17, 1990, when the unit was shut down to MODE 3 as scheduled for routine required surveillances and a special test of a pressurizer snubber which had been overlooked during previous outage testing. The scheduled shutdown concluded 251 consecutive days power production since start up from the last refueling outage in July, 1989. The unit re-entered MODE 1 on March 20, 1990, successfully completing the three day outage as planned.

b. Unit 2 operated routinely at design power levels throughout the inspection period.

c. 2-OHP 4021.032.009, "Aligning DG 2CD Subsystems For Standby Operation." A complete walkdown of the starting air system for the Unit 2 CD Emergency Diesel Generator was performed using the procedure.

The position of all thirty-nine valves listed on the attachment correctly matched those in the field. Some valves were found sealed or locked in position, but this was not specified on the attachment. The inspector was directed to another lineup procedure (2-OHP 4030 STP.035) used to verify proper alignment and securement of controlled valves for the entire unit. All safety related sealed valves (which included those in 2-OHP 4021.032.009) are verified weekly using STP.035.

d. The inspector walked down numerous Unit 1 containment electrical penetrations. Both a general inspection of overall condition and a specific inspection of proper assembly alignment and support, were performed. Another utility was recently discovered to have missing or maladjusted alignment/support bolts in some electrical penetration assemblies. No problems were noted in inspection of about three dozen assemblies for D. C. Cook Unit 1.

e. On one of the routine plant tours with site senior management, the Nuclear Sampling Room was entered and the sample sink door was found open without an attending technician. The Assistant Plant Manager instructed a Radiation Protection Technician outside the area about the sample sink precaution. The door was found properly shut during an independent inspector tour the next day.

f. The inspector accompanied one of the initial entry inspection teams on an inspection tour of the Unit 1 lower containment when the unit was shut down to MODE 3 on March 17, 1990. Among the items inspected were steam generators No. 2 and 3, the incore flux thimble seal table, reactor coolant pump studs, control rod drive housing cooling fans, and general area primary and secondary piping. The licensee members of the inspection team documented and reported all observations for evaluation. During the tour, the inspector verified the presence of special locks to control entry under the reactor vessel.

g. Problem Report (PR) 90-0183, "...the JCC (Job Coverage Coordinator) was notified that the nuclear sampling room CAM (Continuous Air Monitor) was alarming..." All personnel were evacuated and an air sample was obtained. No radioactive iodine was detected and isotopic concentration analyses measured less than imposed limits. The determination of cause was attributed to a localized temperature inversion and the presence of radon gas. Air flow is generally stagnant in the room and the licensee's Corporate Office was requested to evaluate methods of increasing air flow.

No violations, deviations, unresolved or open items were identified.

3. <u>Maintenance (62703, 42700)</u>

Maintenance activities in the plant were routinely inspected, including both corrective maintenance (repairs) and preventive maintenance. Mechanical, electrical, and instrument and control group maintenance activities were included as available.

The focus of the inspection was to assure the maintenance activities reviewed 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. Job Order (JO) A016145: "Disassemble No. 1 Turbine Room Sump Pump and inspect for internal blockage. Repair or replace damaged or unserviceable components and reassemble." This job order was written to investigate shaft vibration and abnormal noise from the sump pump. Upon initial disassembly, water rushed from the pump suction line onto the floor, which is indicative of a leaking suction isolation valve. A plug to hold the leakby was ineffective so the line was sealed with a wood plate and a gasket. A Job Order was written to repair the leaking butterfly-type suction isolation valve, but a dry sump needs to exist to make the proper repair. The maintenance mechanic found a small metallic barrel plug inside the pump which was believed to have entered the sump during housekeeping activities in the turbine building. The small plug badly damaged the pump's impeller and shaft, both of which will be replaced along with new bearings and mechanical seals.

b. During a tour of the Unit 1 residual heat removal heat exchanger rooms, the inspector examined the reach-rod operators for valves 1-RH-116E, 1-RH-116W, 1-RH-128E and 1-RH-128W. Each of these valves is operated by a handwheel from just outside the heat exchanger room. With the exception of one zerk at the handwheel connection itself on valve 1-RH-116W, the various U-joints and gear boxes on these valves all appeared not to have been greased recently. Some zerks were painted over. Also, Job Order tag (No. 713802) dated April 9, 1987, was hung on the handwheel to valve 1-RH-128E, indicating the valve was "hard to operate" and needed attention three years ago. Subsequent followup showed the repair was reported as complete in April 1987.

   The inspector also followed up by reviewing the preventive maintenance records for the periodic lubrication of reach-rod operated valves. The current list includes 184 valves, of which 174 (excluding 10 in extreme high radiation areas) were inspected and lubricated in 1989. Specifically, Job Order No. 758201 covered valves 1-RH-116E and 1-RH-116W (this work was performed in June, 1989) while Job Order No. A004897 covered valves 1-RH-128E and 1-RH-128W and was performed in July, 1989. The inspector's review indicated that the fittings in question were greased according to the licensee's schedule and apparently painted over thereafter.

c. Job Order (JO) B001654: "Reinstall and add additional insulation of Unit 2 main steam stop valves 2-MRV-210 and 2-MRV-240." The inspector observed work on 2-MRV-210 only.

d. On one tour of the Unit 1 West residual heat removal pump room, the inspector observed that the reach-rod operator on a drain valve (1-RH-109W) was partially disassembled and lying on the room floor. No tools or other evidence were present to suggest work in progress. The handwheel outside the room contained a Job Order tag (No. A015990) dated January 18, 1990, which stated the subject drain valve was leaking by. A check with the Maintenance Department showed they had not initiated any work on the valve. These facts made it appear someone may have "repaired" the leak by disconnecting the reach-rod assembly (so as to permit direct access to the valve handle) to shut the valve tighter than the normal reach-rod operator could manage. The Shift Supervisor was informed of the disassembled condition of the operator and he initiated a Job Order.

   The licensee has in place administrative controls to prevent unauthorized maintenance. Lacking specific evidence that this was unauthorized maintenance, this issue was discussed at the management exit.

e. 12 THP 6020.LAB.062: "Miscellaneous Chemical Additions." The inspector observed addition of corrosion-resistant makeup water/fluid to the 2CD emergency diesel generator jacket water cooling system, which had a small chronic leak.

f. **12 MHP 5021.001.075: "Repair Procedure For Fisher Controls Angle Valves." The procedure was used in conjunction with Job Order No. A010718 to repair seat leakby on valve 1-MRV-221 (steam generator stop valve 1-MRV-220 steam cylinder Train A dump valve). The repair was performed while operating the unit in MODE 1, for which a 4-hour Technical Specification ACTION Statement on the associated stop valve was intentionally entered.

   Repairs to the dump valve were deemed necessary partly because of the problems recently encountered with slow stroke times on Unit 2 stop valves (ref. NRC Inspection Report 50-315/90005(DRP); 50-316/90005(DRP)).

g. Job Order (JO) A014996, "Calibrate Nuclear Instrumentation System Power Range Channel With Attached Data." The job was performed in conjunction with **1 IHP 6030 IMP.351, "Power Range Nuclear Instrumentation Calibration - N43." A faulted current meter was discovered for which no spare "Class 30" (nuclear grade) meter was in stock. The licensee removed and installed a meter from a qualified spare drawer as replacement.

h. Licensee corrective action documents were routinely reviewed. One purpose of this review was to identify potential problems in the area of effective implementation of maintenance or construction activities: procedure quality and use; work practices; errors; and similar indicators.

(1) Problem Report 90-0250: A QC hold point for inspection of weld fitup on a conduit support was bypassed, and the support was not drilled with the specified weep holes.

(2) Problem Report 90-0272: Bonnet bolts/studs on safety valve 12-SV-87 were torqued to half-inch torque specifications, but they were only seven-sixteenths studs.

(3) Problem Report 90-0276: Train B heat trace circuits 133-S1 and 132-S2 were found wired in series vice in parallel as specified by design.

(4) Problem Report 90-0277: Valve 2-IMO-314 position monitor light would not indicate; an inspection found several loose terminal connections, including one for the monitor light which had fallen off completely.

(5) Condition Reports 1-02-90-0230 and 1-02-90-0263 (Problem Report 90-0201): Indicated Hydrogen Recombiner temperature discrepancies persisted despite three reported repair attempts and, as discussed in Paragraph 4.c below, the Unit 1 No. 2 Recombiner was ultimately inoperable for over 20 days.

i. Job Order (JO) A015696, "WD 317 is leaking by at a rate of approximately 40 gallons per hour. Please readjust or replace diaphragm." Two job orders were written on valve 12-WD-317 (Chemical Volume and Control System Monitor Tanks' South Transfer Pump Discharge to Monitor Tank No. 3 Shutoff Valve) within a two month period to address the same leakby problem. JO A015677 dated January 10, 1990 described the condition as resulting from a travel stop maladjustment. The diaphragm was noted in good condition, but was replaced as a preventive measure. JO A015696 dated March 2, 1990 described the leakby as resulting from a loose diaphragm "jam nut".

Proper documentation of post maintenance testing was checked. JO package A015677 contained no "Technical Specification and Testing (TST)" form while JO package A015696 did, and stated verification of no leakby. However, the procedure used in conjunction with both JO packages (**12MHP 5021.001.023, "Disassembly, Repair and Reassembly of Grinnel Air-Operated and Handwheel-Operated Diaphragm Valves") does contain appropriate post maintenance testing and acceptance criteria at Step 9.3.

One final observation was that unlike 12-WD-317, valve 12-WD-318 (Monitor Tank No. 4 Shutoff Valve) was noted to have stainless steel cover bolting. A consultation with the Design Change Coordinator produced a March 2, 1989 letter, which provided for deletion of certain valves from the list destined for carbon steel body-to-bonnet stud and nut replacement. Apparently the studs and nuts on 12-WD-318 were changed to stainless steel prior to the effective date of the letter.

No violations, deviations, unresolved or open items were identified.

4. <u>Surveillance (61726, 42700)</u>

The inspector reviewed Technical Specifications required surveillance testing as described below and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that Limiting Conditions for Operation were met, that removal and restoration of the affected components were properly accomplished, that test results conformed with Technical Specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The following activities were inspected:

a. **12 THP 6040 Per.362, "Incore Moveable Detector System Post Maintenance and Functional Test."

b. 12 THP 6040 PER.323, "Flux Mapping System Operation and Supportive Data Collection."

   The above two activities were conducted consecutively on February 23, 1990, to functionally check out and then operate the incore system preparatory to the quarterly incore/excore calibration.

c. **1 OHP 4030 STP.013A, "No. 2 (Train A) Electric Hydrogen Recombiner Semi-Annual Functional Test."

   **12 THP 4030 STP.206.2, "Surveillance Test Procedure - Electric Hydrogen Recombiner No. 2."

   These tests were performed as restoration tests subsequent to recombiner maintenance. The recombiner was deemed inoperable February 8, 1990 because of divergent temperature indication on one of three thermocouples noted during a shift turnover panel walkdown. An investigation led to a plan to replace the defective thermocouple - to be performed in conjunction with a heater element replacement because of the design of the recombiner. The new heater element was taken from several years in storage and tested lower than the administrative insulation resistance value (it was greater than the Technical Specification limit) stated in the subject surveillance procedure. The licensee energized the recombiner to "dry out" the heater, and the recombiner was classified OPERABLE after successful testing on March 1, 1990, approximately 20 days into the 30 day Technical Specification ACTION statement.

d. **12 THP 4030 STP.059, "Sampling Room Area Monitor (R-6) Surveillance Test (Quarterly)."

This was a return-to-service test performed on the nuclear sampling room area radiation monitor after maintenance, rather than a routine quarterly test.

e. **2 IHP 6030 IMP.449, "Power Range Nuclear Instrumentation Calibration N41."

This was a post maintenance test performed on February 20, 1990 following repairs to power range nuclear instrument N41, the indication for which failed downscale earlier that day. At the time the inspector observed this activity, he also reviewed the logbook used to record usage of calibrated measurement and test equipment. The log had not yet been updated to show use of certain equipment for this calibration. The following day, after calibration was complete, the inspector reexamined the logs and found they had been properly updated.

f. **1 IHP 4030 STP.021, "Steam Generator 1 and 3 Mismatch Protection Channel II Surveillance (Monthly)."

The annual calibration for one of the voltmeters used during this surveillance was due two days later. The technicians doing the surveillance were aware of the due date.

g. **1 IHP 6030 IMP.301, "Steam Generator 3 and 4 Mismatch Protection Set I Transmitter Calibration."

h. **12 THP 6030 IMP.012, "Radiation Monitoring System Calibration Air-Liquid-Gas." The calibration was observed for the Unit 2 West Essential Service Water header liquid monitor. No problems were observed.

No violations, deviations, unresolved or open items were identified.

5. <u>Engineering and Technical Support (71707, 90713)</u>

The inspector monitored engineering and technical support activities at the site and, on occasion, as provided to the site from the corporate office. The purpose of this monitoring was to assess the adequacy of these functions in contributing properly to other functions such as operations, maintenance, testing, training, fire protection and configuration management.

a. During one tour in the 1CD emergency diesel generator room, the inspector observed workers engaged in installation of conduit and cable through the room overhead as part of design change RFC-2950.

The design change is to enhance radio communications capability on the in-plant radio system utilized by plant operators. The conduit/cable simply passes through the room from one end to the other.

The job was being performed off a large scaffold extending the length of the room, to a height of about ten feet or greater, with much of it parallel and immediately adjacent to the emergency diesel engine and generator. The diesel generator was considered OPERABLE at the time. The inspector observed the scaffold was held in place by cables to pad-eyes in the room floor. The inspector followed up with the design change coordinator's group to ascertain the seismic design and controls on this scaffold, if any, and to get information on generic seismic considerations for scaffold erected in safety related areas. An NRC Region III specialist will be asked to review such information.

b. An NRC Region III specialist inspector reviewed the licensee's "Reactor Containment Building Integrated Leakage Rate" reports for Units 1 and 2 submitted to the NRC on September 8, 1989, for Unit 2, and October 13, 1989, for Unit 1. The inspector determined that it accurately reports the leakage rates and events regarding both Units' Type A tests performed in 1989. However, the inspector noted that for some penetrations (Component Cooling Water (CCW) for one) the licensee took no penalty for repairs and adjustments (R&A) made prior to the CILRT on the basis that they were part of a closed loop inside containment. The only given justification was that the system is Seismic Class 1. By definition, a closed loop system must meet other requirements such as safety class 2, missile and pipe whip protection, etc. From conversations with the licensee, it appears that the system is safety class 2 and probably meets all other criteria for a closed loop system.

This issue again highlights the fact that the licensee's Updated Safety Analysis Report (USAR) Table 5.4-1 does not address the requirements for Type B and C testing of containment penetrations. During a previous inspection (Inspection Report No. 50-315/89007(DRS); 50-316/89007(DRS)) this issue was discussed with the licensee and it was suggested that a review of all containment penetrations, and updating of Table 5.4-1 be performed. The review should justify the positions taken in the updated Table 5.4-1 and should be submitted to the NRC Office of Nuclear Reactor Regulation (NRR) for review and concurrence. The licensee indicated a willingness to perform such a study. Pending completion of the review and acceptance by NRR, the issue of whether the CCW system can be considered a closed loop inside containment - and consequently whether the containment "as found" results were satisfactory, is considered an Unresolved Item (316/90006-01).

One unresolved item and no violations, deviations, or open items were identified.

6. Safety Assessment/Quality Verification (92700, 92720)

The effectiveness of management controls, verification and oversight activities, in the conduct of jobs observed during this inspection, was evaluated.

The inspector frequently attended management and supervisory meetings involving plant status and plans and focusing on proper co-ordination among Departments.

The results of licensee auditing and corrective action programs were routinely monitored by attendance at Problem Assessment Group (PAG) meetings and by review of Condition Reports, Problem Reports, Radiological Deficiency Reports, and security incident reports. As applicable, corrective action program documents were forwarded to NRC Region III technical specialists for information and possible followup evaluation.

7. Emergency Preparedness (82201, 82203)

a. The licensee declared an Emergency Plan "Unusual Event" on February 14, 1990 when a small leak was discovered on a Unit 2 pressurizer sample line fitting outside the containment but inboard of the first containment isolation valve. The licensee attempted to stop the leak by tightening the fitting and closing an in-line control valve (CV) inside containment. Neither was completely successful, as the fitting continued to leak at about 2 cc/min.

   The presumption that containment leakage limits were not being met led to application of the associated Technical Specification Limiting Condition for Operation (LCO). The LCO called for unit shutdown within four hours.

   Subsequently, the Technical Engineering group determined by calculation that the observed leakrate was "negligible" compared to containment leakrate limits and the Unusual Event was terminated and the shutdown aborted with the unit at about 80-percent power. Ascension to 100-percent power began and was reached later that day.

b. The licensee activated the onsite fire brigade on February 23, 1990, in response to the discovery of a small fire in a heating boiler exhaust penetration in the turbine building. The area had recently been modified into offices and a new seal installed around the exhaust stack penetration (approximate 4-foot diameter pipe). After running the heating boiler for several hours the area around the seal ignited. The licensee's determination of cause was misapplication of the seal material. The seal material was installed around an uninsulated portion of the pipe.

No violations, deviations, unresolved or open items were identified.

8. <u>Reportable Events (90712, 92700)</u>

The following Licensee Event Reports (LERs) were reviewed to verify compliance to reporting requirements and, as applicable, accomplishment of immediate corrective action. Further, both items were subjected to a technical review by an NRC Region III fire protection specialist who, in concert with the resident inspector, joined in the determination that appropriate measures had been accomplished to correct and prevent recurrence of the items.

a. (Closed) Licensee Event Report LER 315/89013: Required Compensatory Action Not Completed. During the period from September 6 through October 3, 1989, the East Motor driven Auxiliary Feedwater Pump (MDAFP) room door (a fire barrier penetration door) was secured in the open position due to a faulty latch. A roving firewatch toured the area each 30 minutes, so fire protection requirements were met; however, the door also acts as protection of the MDAFP from postulated high energy line breaks in the corridor outside or in the adjacent turbine-driven pump room. For this purpose, the licensee's administrative controls required a continuous watch on this particular door. This was not provided. The licensee changed the administrative controls to apply the Technical Specification LCO (72-hour limit for an "inoperable" MDAFP) in any future similar occurrence.

b. (Closed) Licensee Event Report LER 315/89015: Required Compensatory Action Not Completed. In this event, an inoperable detection circuit required assignment of a fire watch to the affected area within an hour, to inspect the zone at least hourly thereafter. The detection circuit was "inoperable" due to a continuous alarm being caused by planned welding in the area. Due to an error on the part of control room personnel, the fire watch was established in a different fire zone where other ongoing activities were expected to affect fire detection circuits but, in fact, did not. The condition lasted just over two hours. Automatic fire suppression remained operable throughout this time. Subsequently, each of the control room personnel involved was counseled regarding accurate communications and panel assessment. All were well aware of their responsibilities in these areas. No previous similar events had occurred.

Both of the above items involved failure to implement Technical Specification and/or procedure requirements and are considered violations. In each case, the root cause of the problem was a miscommunication concerning status details on fire protection equipment. This led in turn to application of the wrong administrative and compensatory measures (8.a above) or application of correct measures to the wrong area (8.b above). These violations, however, met the tests of 10 CFR 2, Appendix C, Section V.A; consequently, no Notice of Violation will be issued and these matters are considered closed.

Two violations (not cited) and no deviations, unresolved or open items were identified.

9. <u>NRC Compliance Bulletins, Notices and Generic Letters (92703)</u>

The inspector reviewed the NRC communications listed below and verified that: the licensee has received the correspondence; the correspondence was reviewed by appropriate management representatives; a written response was submitted if required; and, plant-specific actions were taken as described in the licensee's response.

a. (Closed) NRC Bulletin No. 88-03, "Inadequate Latch Engagement In HFA Type Latching Relays Manufactured By General Electric (GE) Company." The licensee has notified the NRC that the relays installed in safety related systems were not HFA type latching relays manufactured by G.E. This item is considered closed.

b. (Closed) NRC Bulletin No. 88-04, "Potential Safety Related Pump Loss." The licensee has committed to the short term corrective actions stated in NRC Inspection Report 50-315/89023(DRP); 50-316/89023(DRP), Paragraph 11.b by letter dated January 5, 1990 from J. R. Sampson to Operation Shift Supervisors. This item is considered closed.

c. (Closed) NRC Bulletin 88-05, "Noncomforming Materials Supplied by Piping Suppliers, Inc., at Folsom, New Jersey, and West Jersey Manufacturing Company, at Williamstown, New Jersey." This item is considered closed based on the information stated in NRC Inspection Report 50-315/88020(DRP); 50-316/88023(DRP) that all items provided by the respective companies have been verified to meet the quality requirements for the item.

d. (Closed) The following Bulletins are being administratively closed. No additional inspections beyond the general review described above, has been scheduled or will be performed.

NRC Bulletin No. 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes."

NRC Bulletin No. 88-06, "Action To Be Taken For The Transportation Of Model No. Spec 2-7 Radiographic Exposure Device."

NRC Bulletin No. 88-07, "Power Oscillations in Boiling Water Reactors (BWRS)."

NRC Bulletin No. 88-08, "Thermal Stresses In Piping Connected To Reactor Coolant Systems."

NRC Bulletin No. 88-09, "Thimble Tube Thinning in Westinghouse Reactors."

e. (Closed) Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary components in PWR Plants." The licensee's boric acid corrosion prevention program was audited by NRC in July 1989 and, as documented via letter (J. G. Giitter to M. P. Alexich dated February 22, 1990) and attached trip report, the audit findings were acceptable.

No violations, deviations, unresolved or open items were identified.

10. Unresolved Items

Unresolved Items are matters about which more information is required in order to ascertain whether they are acceptable items, violations, or deviations. An Unresolved Item disclosed during the inspection is discussed in Paragraph 5.b.

11. Management Meeting (30702)

A management meeting, attended as indicated in Paragraph 1.a, was conducted at the D. C. Cook plant site on March 15, 1990. The focus of the meeting was on maintenance issues. The specific topics upon which the licensee made presentations were: management commitment and involvement in maintenance; the preventive maintenance program; maintenance effectiveness and monitoring; technical and engineering support of maintenance; and, plant material condition. There was a free exchange of questions, answers and general discussions in each topical area.

12. Management Interview (30703)

The inspectors met with licensee representatives (denoted in Paragraph 1) on March 21, 1990, to discuss the scope and findings of the inspection as described in these "Details". 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/processes as proprietary.