U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-315/85010(DRP); 50-316/85010(DRP)

Docket Nos. 50-315; 50-316 Licenses No. DPR-58; DPR-74

Licensee: American Electric Power Service Corporation
Indiana and Michigan Electric Company
Columbus, OH 43216

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

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

Inspection Conducted: March 19, 1985 through April 22, 1985

Inspectors: B. L. Jorgensen

J. K. Heller

C. L. Wolfsen

Approved By: for John F. Suermann G. C. Wright, Chief
Reactor Projects Section 2A

May 3, 1985
Date

Inspection Summary

Inspection on March 19, 1985 through April 22, 1985
(Reports No. 50-315/85010(DRP); 50-316/85010(DRP))
Areas Inspected: Routine unannounced inspection by the resident inspectors of licensee actions on previous inspection findings; operational safety; maintenance; surveillance; Licensee Event Reports; IE Bulletins; design changes and modifications; and, independent inspection areas. The inspection involved a total of 204 inspector-hours by three NRC inspectors including 31 inspector-hours off-shift.
Results: No violations or deviations were identified in any of the areas inspected.

DETAILS

1. Persons Contacted

   * W. G. Smith, Jr., Plant Manager
   * B. Svensson, Assistant Plant Manager
   * T. Kriesel, Technical Superintendent-Physical Science
   * A. Blind, Assistant Plant Manager
   * K. Baker, Operation's Superintendent
   N. Williams, Maintenance Superintendent
   * J. Stietzel, Quality Control Superintendent
   T. Beilman, Quality Assurance Superintendent

   The inspector also contacted a number of licensee and contract employees and informally interviewed operation, technical, and maintenance personnel during this period.

   *Denotes personnel attending exit interview on April 26, 1985.

2. Licensee Actions on Previously Identified Items

   a. (Open) Open Inspection Items 315/83-14-03 and 316/83-15-02: Incompletely documented design changes were approved for installation and use. Review of design changes during this inspection (see Paragraph 8.b.(1)) identified another case analogous to that on which this open item is based. Specific detailed review of this matter by NRC Region III specialists will be arranged for a future inspection.

   b. (Closed) Noncompliance Item 315/82-21-05: Failure to perform PNSRC review on RFC 12-2500 prior to implementation. Several RFC packages reviewed during this inspection were found to properly reflect prior PNSRC review and to contain detailed documentation of phone conversations among involved personnel, lack of which contributed to the problem identified in this item.

   c. (Closed) Open Item 315/82-21-06 and 316/82-24-07: Replacement of non-essential service water check valves. This was accomplished under RFC 12-2549, which applied to both units and was completed in late 1982 and early 1983 for Units 1 and 2 respectively.

   d. (Closed) Open Item 315/81-24-04 and 316/81-26-05: Resolution of ice condenser door freezing problems. The licensee initiated two RFC's (12-1758 involving a programmable timed defrost cycle and 12-1791 involving insulation coating the top deck door dividers) to address this problem, which had resulted in several Licensee Event Reports (LER's). These RFC's were installed by early 1984 and have proven effective in controlling the problem.

e. (Closed) Open Item 315/82-04-05 and 316/82-04-05: Review of revised PNSRC safety review checklist and subcommittee functioning. The revised safety review checklist has been implemented effective with Revision 3 of PMI-1040 dated October 1, 1984. PNSRC subcommittee functions have gradually expanded with increased experience and are now considered to provide a positive contribution to committee functioning.

No violations or deviations were identified.

3. Operational Safety Verification

The inspector observed control room operations including manning, shift turnover, approved procedures and LCO adherence, and reviewed applicable logs and conducted discussions with control room operators during the inspection period of March 19 through April 22, 1985. Observations of control room monitors, indicators, and recorders were made to verify the operability of emergency systems, radiation monitoring systems, and nuclear and reactor protection systems. Reviews of surveillance, equipment condition, and tagout logs were conducted. Proper return to service of selected components was verified. Tours of the auxiliary building, turbine building, and screenhouse were made to observe accessible equipment conditions, including fluid leaks, potential fire hazards, and control of activities in progress.

By observation and direct interview it was verified that the physical security plan was being implemented in accordance with the station security plan. The inspector did notice that guards, while at a duty post, were reading material that did not appear to be job related. This was discussed with licensee management who committed to review the matter and strengthen administrative controls if required.

At about 0840 on April 3, 1985 the Unit 2 off gas monitor on the steam-jet air ejector alarmed, indicating apparent commencement of a small primary-to-secondary leak. Readings subsequently stabilized at a new elevated level equivalent to a fraction (about one-fifth) of the Technical Specification limit of 500 gallons per day. The inspector observed activities related to monitor trending and specific identification of the affected steam generator (No. 23), and discussed these matters with Operations and Technical-Physical Sciences personnel. The licensee has continued to monitor the situation closely, but as of the end of this inspection period, no further changes had occurred.

The licensee declared an Emergency Plan "Unusual Event" on April 2, 1985 when both emergency diesel generators were concurrently "inoperable" in Unit 2. The 2CD diesel had been taken out of service for maintenance and the 2AB diesel was started to verify operability. When 2AB was shut down, it continued to run (unloaded) because a stuck fuel pump linkage on one cylinder injector allowed continued fuel feed to that cylinder. Diesel 2AB was manually secured, declared "inoperable", and the Unusual Event was declared. The inspector observed control room activities relating to restoration of diesel 2CD, which was accomplished in less than one hour and in compliance with the applicable Technical Specification LCO. This action terminated the "Unusual Event".

The inspector reviewed the administrative controls for Condition Reports (PMI-7030), which are used to initiate Licensee Event Reports. PMI-7030, section - "Reportability" at Paragraph 4, states that..."a Condition Report is not required for voluntary entry into Technical Specification Action Statements provided the PNSRC approved the entry and the Technical Specification Action Statements are not exceeded." The inspector informed the licensee that if they enter the Action Statement of Technical Specification 3.0.3 ("Motherhood Statement") a Licensee Event Report is required. (Reference: Paragraph 2.4 of NUREG 1022 - Supplement 1).

During this inspection, licensee control of selected power operated valves was examined. Unit 1 Technical Specification 4.5.2.a.4 requires (on a 31 day frequency) that the listed valves be verified in the specified position with control power locked-out (emphasis added), whereas Unit 2 Technical Specification 4.5.2.a requires (on a 12 hour frequency) that the analogous valves be verified in the specified position with power to the valve operator removed (emphasis added). The surveillance procedures for both units required the operator to verify that the control power is "locked-out". The inspector questioned whether verifying that the control power is "locked-out" complies with the Unit 2 Technical Specification. Modifications to original design were made in both units, consisting of: (1) a key-lock feature for the control switch, (2) separate control power locked-out switch, (3) annunciation of control power locked-out, and (4) valve position indication when valve control power is deenergized. The licensee concluded that Unit 2 Technical Specification 4.5.2.a is met when the operator verifies control power is locked-out. The inspector reviewed the electrical drawings and the Safety Evaluation by the Office of Nuclear Reactor Regulation supporting Ammendment Number 18, which changed Unit 1 Technical Specification 4.5.2.a.4 to its current revision. It appears the intent of Unit 2 Technical Specification is met when the operators verify that control power is locked-out, but it is not clear why the two Technical Specifications are different. This was discussed at the exit interview with a request that a formal review be performed and, if appropriate, Technical Specification revisions be requested. (Open Item 316/85010-01).

No violations or deviations were identified.

4. Monthly Maintenance Observation

Station maintenance activities of safety related systems and components listed below were observed and/or reviewed to ascertain that they 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.

The following maintenance activities were observed and/or reviewed:

Instrument calibration check: WFI 721 - essential service water flow from EDG

JO Nos. 13586 through 13589: implement RFC 12-2530 (check valve installations) for reactor coolant pumps 21 through 24

JO Nos. 38608 through 38611: replace incorrect check valves installed under above JO Nos. 13586-13589.

JO Nos. 75508 and 040026: replace upper and lower PORV limit switch gaskets (Unit 1 and 2 respectively) per RFC 12-2728

JO Nos. 034727 through 034730; 034742 through 034744; and 040507: implement and/or support implementation of RFC 02-2784 (replacement of Unit 2 AB station battery)

JO Nos. 034740 and 034741: pre-op test new AB station battery

JO Nos. 75629 through 75635 (Unit 1) and 035286 through 035292 (Unit 2): replace 100 amp pressurizer heater breakers with 150 amp breakers per RFC 12-2632

No violations or deviations were identified.

5. Monthly Surveillance Observation

The inspector reviewed Technical Specifications required surveillance testing on the systems listed 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 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.

**1 THP 4030 STP.019	Steam Generator 1 and 2 Mismatch Protection Set 1
**2 OHP 4030 STP.005	Emergency Core Cooling System Operability Test (Train A test for Unit 2)
**1 OHP 4030 STP.005	Emergency Core Cooling System Operability Test ("N" Safety Injection pump only - Unit 1)
**1 THP 4030 STP.06	Overpressure and Overtemperature Protection Set III Monthly Surveillance

**2 THP 4030 STP.154	Control Room Area Monitor 2R-1 Quarterly Surveillance
1 OHP 4030 STP.018	Main Steam Line Isolation Valve Operability Surveillance Testing

Step 3.1 of STP.018 requires full closure testing of the MSIV while the plant is in Hot Standby with Tavg > 541 F, or in Mode 1 or 2. Unit 1 Technical Specification 4.7.2.5.b requires full closure testing while in Hot Standby with Tavg > 541 F. It appears the potential exists to test the valves in a plant mode not allowed by the Technical Specification. The inspector reviewed the completed files of STP.018 and verified that for 1984 the testing was performed while in Hot Standby with Tavg > 541 F. STP.018 was discussed with the Operations Department who agreed to make the wording of Step 3.1 comply with Technical Specification 4.7.2.5.b.

No violations or deviations were identified.

6. Licensee Event Reports

Through direct observation, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence had been accomplished in accordance with Technical Specifications. The following LER's are considered closed:

Unit 1

RO 85-003-0	A Train battery and B Train emergency diesel were concurrently inoperable during Mode 5. Technical Specifications requiring establishment of containment integrity within 8 hours were not adhered to, as the condition was not recognized within the eight hour period. This constitutes a violation for which, in accordance with the NRC Enforcement Policy (identified, reported, and corrected by the licensee and not Severity Level III or greater), no Notice of Violation is being issued. Appropriate corrective action has been implemented.

Unit 2

RO 85-001-0	An automatic ESF actuation (auxiliary feedwater automatic start) occurred due to imprecise steam generator water level control in Mode 3. Equipment functioned as designed.
RO 85-002-0	An automatic ESF actuation (reactor trip on steam generator low-low level) occurred due to a steam flow transient initiated by unisolating a steam dump valve while in Mode 2 and manual level control. Equipment responded as designed. An Operating Memo was issued

to all crews to familiarize them with the circumstances of this event.

RO 85-003-0 Reactor trip on loss of CRID III. The turbine driven auxiliary feedwater pump (TDAFP) failed to start automatically, due to excessive clearance in the trip and throttle valve latching mechanism. Shiftwise verifications of valve latching (both Units in appropriate Modes) are being implemented, as is weekly start testing.

Both RO 85-002-0 and 85-003-0 were previously reviewed in part in IE Inspection Report 316/84-25(DRP).

One violation (not cited, as described above) and no deviations were identified in this area.

7. IE Bulletins

IE Bulletin 80-24, "Prevention of Damage Due to Water Leakage Inside Containment", specified information to be provided and actions to be taken by licensees to address concerns identified via a containment flooding event at Indian Point Unit 2 in October 1980. The licensee responded by letter (AEP:NRC:00499) dated January 21, 1981. Initial NRC evaluation of the response identified incomplete information was provided to give adequate assurance the instrumentation and pumping systems installed would be maintained in good operating condition and that any failure of these components would be detected in a timely manner. Subsequent followup with the licensee concerning these matters led to licensee commitments to perform periodic (bi-monthly) visual inspections of the containment pipe tunnel sump; to implement periodic (18 month) surveillance of installed water detection and removal equipment; and to add (via RFC 12-2451) additional control room water level monitoring instrumentation. These actions were deemed appropriate to satisfy NRC concerns, and followup verification was planned to support Bulletin closeout.

The additional instrumentation of RFC 12-2451 has been installed. The surveillance activities verifying continued operability of water detection and removal equipment have been incorporated into approved surveillance procedures and are ongoing. The periodic containment entries to perform visual inspections have been ongoing. The licensee has requested by letter (AEP:NRC:00499A) dated March 25, 1985 that the visual inspection entries be terminated effective May 1, 1985. Based on verification of the availability and reliability of installed equipment as discussed above, termination of the visual inspections is considered acceptable to the resident inspector. This information will be considered in the pending Region III response to the licensee's March 25, 1985 letter. The inspector has no further questions concerning these matters, and this Bulletin is considered "closed".

No violations or deviations were identified.

8. Design Changes and Modifications

The inspector reviewed selected design change control procedures and Request for Change (RFC) packages to verify: that changes were made in accordance with 10 CFR 50.59; that they were reviewed in accordance with Technical Specifications and the established Quality Assurance program; and, that the changes were conducted in accordance with written procedures which included appropriate inspections, tests and acceptance values or standards. Associated test records were reviewed to verify acceptable performance of modified equipment. The inspector verified applicable procedures and drawings were changed to reflect the modifications, and design change records were being maintained as described in 10 CFR 50.59 and the QA program.

a. Procedures

(1) PMI-5010 "Maintenance, Repair and Modification Policy"

(2) PMI-5040 "Design Change Control Program"

(3) PMI-5075 "Repairs, Replacements and Alterations of ASME Code Class Components and Systems"

(4) NSL/7 "Safety Review of Design Changes" (AEP:SC Nuclear Safety and Licensing Section procedure)

(5) DCC QA 103 QCN "N" List

b. RFC Packages

(1) RFC 12-2728: replacement of upper and lower PORV limit switch gaskets with "improved" type. The Safety Review Memorandum supports the modification as a means of extending the environmentally qualified life of the instrument. Installation was subsequently authorized, however, despite an RFC "open" item involving incomplete documentation of environmental qualification of the modified assemblies as to aging. This necessitated a waiver of existing procedural controls requiring closeout of RFC "open" items prior to implementation. This is analogous to a previously identified "Open Item" (315/83-14-03; 316/83-15-02) involving incompletely documented design changes approved for installation and use (see also Paragraph 2.a).

(2) RFC 02-2784: replacement of Unit 2 AB battery. The original Exide lead-antimony battery of 120 cells was replaced with a C and D Battery Co. lead-calcium battery of 116 cells when the original failed a service-life test. A minor discrepancy in this activity occurred when the new battery was released to the Operations Department, and returned to service after post-installation testing, without notification to and concurrence from the assigned Design Change Coordinator. The final post-installation testing was performed using the standard 18 month

surveillance test procedure, which properly concludes with instructions to notify Operations to return the system to normal operating conditions.

(3) RFC 12-2632: replacement of 100-amp pressurizer heater circuit breakers with 150-amp breakers (including selected associated cable replacement) to reduce unnecessary breaker trips. The design change was designated "non-safety related" but seismic qualification requirements were stipulated and met, and Quality Assurance was properly invoked in purchase documentation and in pre-operational calibration and testing.

(4) RFC 12-2530: installation of check valves in reactor coolant pump No. 1 seal leak off lines to prevent reverse flow into the seal at low reactor coolant system pressure. Velan Co. 1500 psi, 2-inch, stainless steel, swing check valves were specified and procured specifically for this purpose (Purchase Order 2003-041-1), but the contractor performing the work installed Conval piston-type check valves instead, in all four Unit 2 pump leak off lines. The error was identified by the Design Change Coordinator during post-installation inspection (e.g. the system was not returned to service) and the incorrect valves were removed and the correct valves installed. The finding was documented on Condition Report No. 2-04-84-0493, which appears to have been "closed" primarily on the basis of correcting the error (see IE Inspection Reports No. 315/85007(DRS) and No. 316/85007(DRS)). There appears to be a weakness in the licensee's controls in this area, involving the use of drawings and generic component descriptions to specify parts to be installed. In this case, "N" list designator ME2 VKC Y-200C58 applies to 2-inch, 1500 psi, stainless steel check valves, and does not distinguish by manufacturer (Velan/Conval) or by design (swing-type/piston-type). Another Condition Report (12-12-83-1342) had been written just four months previously when swing-type check valves were installed in emergency diesel generator fuel oil lines instead of the intended piston-type check valves. A further complication is probably introduced by the licensee's use of more than one warehousing location for storage and disposal of safety-related parts. The incorrect valves used in initial implementation of the subject RFC were obtained from one warehouse, while the correct valves were waiting in another.

The licensee's system for identification and control of parts, as evidenced by the above, does not assure the use of incorrect parts will be prevented. This is contrary to the requirements of 10 CFR 50, Appendix B, Criterion VIII, and is considered a violation. In that the licensee himself identified this matter, and since a separate Notice of Violation is being issued in the previously referenced inspection report relating to inadequate corrective actions for the identified problem, no Notice of Violation is being issued concerning the problem itself. The licensee is expected to address program compliance to Appendix B, Criterion VIII, in his answer to Inspection Report 315/85007(DRS); 316/85007(DRS). This was specifically discussed at the Management Interview.

No other violations or deviations were identified.

9. Independent Inspection Areas

a. Recently there have been two cases in Region III where Reactor Operators and Senior Reactor Operators did not comply with 10 CFR 50.54(k): An operator or senior operator shall be present at the controls at all times during operation. The Region requested the Resident Inspector to review the administrative procedures and verify that directions or procedures exist that clearly define "at the controls" and verify a directive or procedure exists implementing the requirements for an operator and senior operator to be present in the control room at all times. Donald C. Cook Operations Head Instructions No. 4011 clearly defines these staffing requirements as they apply to the D. C. Cook Plant, and satisfactorily complies with 10 CFR 50.54(k), and 10 CFR 50.54(m)(iii). When NRC Region III Operator Licensing examiners expressed concern that not all licensed operator candidates thoroughly understood plant requirements in this area, during a meeting held on February 28, 1985, the licensee issued an Operating Memo that same day stipulating review of the requirements by all licensed personnel.

b. During recovery from "loss of off-site power testing" a plant in Region III found that the Main Steam Isolation Valves (MSIV) did not close within the required stroke time (< 5 seconds). Review of the event identified an equipment and testing deficiency. Region III requested each Resident Inspector to review the installed configuration and methodology used by the respective licensees to demonstrate operability of the MSIV. The inspector provided the requested information to Region III and concluded that the problem does not exist at D. C. Cook.

c. IE Bulletin 84-03 "Refueling Cavity Water Seal" addresses a refueling cavity seal failure at the Haddam Neck Plant on August 21, 1984. The NRC Region III Division of Reactor Safety (DRS) has responsibility for review and closeout of the technical issues associated with this Bulletin. During this inspection, the resident inspector collected and provided information to support Region III DRS review of this matter.

No violations or deviations were identified.

10. Open Items

Open Items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. An open item disclosed during the inspection is discussed in Paragraph 3.

11. Management Interview

A management interview (attended as indicated in Paragraph 1) was conducted following completion of the inspection. The following matters were discussed:

a. Technical Specification inconsistencies between the two Units as related to selected locked valve controls were noted by the inspector. The licensee was requested to review the matter and request appropriate changes (Paragraph 4).

b. The Licensee Event Reports (Paragraph 6) and previously identified items (Paragraph 2) to be closed on the basis of this inspection were specifically identified.

c. The inspector discussed apparent licensee program deficiencies concerning controls to assure incorrect parts will not be used, and specifically requested the licensee to address corrective actions planned for these deficiencies in his response to a related violation cited in pending Inspection Report 315/85007(DRS); 316/85007(DRS). The licensee agreed to comply with this request (Paragraph 8).

The inspector asked those in attendance at the meeting whether they considered any of the matters discussed to contain proprietary information or other information exempt from disclosure. No such information was identified.