U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-315/88017(DRP); 50-316/88020(DRP)

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

Licensee: American Electric Power Service Corporation
Indiana and Michigan Electric 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: June 14 through July 25, 1988

Inspectors: B. L. Jorgensen
J. K. Heller

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

8/10/88
Date

Inspection Summary

Inspection on June 14 through July 25, 1988 (Reports No. 50-315/88017(DRP); 50-316/88020(DRP))

Areas Inspected: Routine unannounced inspection by the resident inspectors of: actions on previously identified items; plant operations; radiological controls; maintenance; surveillance; fire protection and cleanliness; security; outages; quality program activities; reportable events; and Bulletins, Notices and Generic Letters. The following Safety Issues Management System (SIMS) items were reviewed, with the indicated results: (Closed) Item GSI-22, Generic Letter 85005.

Results: Of the eleven areas inspected, no violations or deviations were identified in any areas. The inspection disclosed no notable strengths or weaknesses in the licensee's activities inspected.

DETAILS

1. Persons Contacted

W. Smith, Jr., Plant Manager
*A. Blind, Assistant Plant Manager, Administration
*J. Rutkowski, Assistant Plant Manager, Production
*L. Gibson, Assistant Plant Manager, Technical Support
*B. Svensson, Licensing Activity Coordinator
K. Baker, Operations Superintendent
J. Sampson, Safety and Assessment Superintendent
*E. Morse, Quality Control Supervisor
T. Beilman, I&C/Planning Superintendent
*J. Droste, Maintenance Superintendent
T. Postlewait, Technical Superintendent, Engineering
L. Matthias, Administrative Superintendent
*M. Horvath, Quality Assurance Supervisor
*D. Loope, Radiation Protection Supervisor
*H. Runser, Production Supervisor
*M. Barfelz, Safety and Assessment 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 Management Interview on July 27, 1988.

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

a. (Closed) Violation (315/84016-02; 316/84018-02): examples of failure to implement timely or effective corrective actions. The examples involved: QC inspector certification (-02A); overdue corrective action responses (-02B); and failure to document and process a QA-identified procedure violation to ensure corrective action (-02C). Corrective actions Items -02B and -02C proved to be relatively straightforward and is described in the licensee letter (AEP:NRC:0911) dated November 28, 1984. Item -02A, however, proved less tractable. A series of letters established progressively later completion dates, and culminated with completion at the end of 1987. At about the same time, however, the licensee informed NRC (letter AEP:NRC:0970A dated December 23, 1987) that certain inspections, which had been overseen by QC inspectors for several years, were going to revert to a "peer" inspection process with optional QC oversight. Subsequent inspection visits, exchanges of correspondence, and a management meeting have culminated in closure of the issue of inspection personnel qualification, authority, and organizational freedom via NRC letter (Miller to Alexich) dated June 21, 1988. The effectiveness of licensee implementation of his programs will continue to be routinely monitored.

b. (Closed) Generic Letter (315/85005-HH; 316/85005-HH): Inadvertent Boron Dilution Events. This letter informed licensees of the NRC staff position on Generic Issue 22 (same subject - also tracked as SIMS Item No. GSI-22) and concluded operating plants like D. C. Cook Units 1 and 2 would not be backfit, but assurance of adequate protection against such events was expected. The inspector reviewed the following licensee procedures addressing this matter:

- 1-(or 2-) OHP 4022.005.001 "Malfunction of CVCS Makeup control"
- 1-(or 2-) OHP 4022.005.002 "Emergency Boration"

The first procedure above contains explicit sections to address symptoms (Paragraph C.3.0), immediate automatic and manual actions (Paragraph C.4.0), and subsequent actions (Paragraph C.5.0) - including use of the second procedure when appropriate - should inadvertent boron dilution occur.

c. (Closed) Generic Letter (315/85013-HH; 316/85013-HH): Transmittal of NUREG-1154 - Davis-Besse Loss of Main and Auxiliary Feedwater. The Generic letter requested only that licensees review NRC findings contained in NUREG-1154 for applicability at their facilities. This has been done. Significant design differences between the respective plants resulted in there being no hardware changes necessitated at D. C. Cook.

d. (Closed) Violation (315/85034-01; 316/85034-01): licensee activities involving instrument calibration documentation, tracking and correction of instrument "drift", and use of derived setpoints each showed instances of noncompliance to applicable procedures. Corrective and preventive actions, as described in the licensee's letter (AEP:NRC:0984) dated April 11, 1986, have been verified. There has been no known recurrence of these examples.

e. (Closed) Generic Letter (315/86007-HH; 316/86007-HH): Transmittal of NUREG-1190 - San Onofre Unit 1 Loss of Power and Water Hammer Event. The Generic Letter requested licensee review of the referenced NUREG and appropriate feedback to plant staff. This has been done.

f. (Closed) Open Item (315/86035-01): Boron-10 depletion in the primary coolant. As updated in Inspection Reports No. 315/87009(DRP); No. 316/87009(DRP) the only remaining attribute for this item was licensee action to assure Boron-10 depletion affects are accounted for as necessary for maintaining adequate Shutdown Margin. The magnitude of the effect was evaluated. This evaluation and PWR industry experience show the effect can be expected to be bounded below 1000 percent milli-Rho (pcm). This value was compared to the control rod worths and position limits for flux shaping and peaking factors during normal power operation. Insertion limit alarm settings are such that operators would receive an alarm and be able to take corrective action before Shutdown Margin problems could develop. For other operating conditions, the boron concentration limit curve in the Technical Data Book (Figure 4.3) has been

adjusted to include an additional 100 ppm margin (equivalent to about 1000 pcm) for Boron-10 depletion. Finally, the licensee monitors for depletion monthly by comparing actual vs. predicted core performance via procedure **12 THP 4030 STP.308, "Boron Curve Update".

g. (Closed) Open Item (315/87017-03; 316/87017-03): TDAFW pump turbine casing drains not isolated from possible steam backflow from other, non-safety systems. The licensee reviewed system design against the possibility that steam backflow from other systems could adversely affect a turbine driven auxiliary feedwater pump. The other systems were found to be tied in via 1-inch and smaller lines, to a common drain pot header, which drains to a miscellaneous drain tank with a 10-inch vent. Significant/adverse steam backflow and back pressure are not considered likely given the size of the various connections and the fact they vent to atmospheric pressure.

h. (Closed) Unresolved Item (316/87023-01): loss of configuration control during original construction resulted in containment divider hatch covers being installed not per design. When a reactor coolant pump hatch cover bolt broke while being torqued on August 24, 1987, a licensee investigation determined numerous bolts in both Units had been cut off at some previous time, moved slightly, and rewelded. Substantial analytical efforts followed, which NRC has previously reviewed, to establish that the "as found" condition met the FSAR design basis. Further, a redesign (also previously reviewed by NRC) was implemented to upgrade the bolting and to reconcile design drawings and records to accurately reflect the as-built condition.

Thus, the bolting proved "strong enough" as found, but was left even stronger.

In-place testing, some of which was witnessed by NRC inspectors, helped prove the cut/rewelded bolts met the design basis for strength. They were not, however, installed in accordance with design drawings. This means design control requirements of 10 CFR 50 Appendix B, Criterion III were violated, as were requirements of Criterion V for accomplishing activities in accordance with applicable prescribed drawings and, possibly, requirements of Criterion X for inspecting and verifying conformance with instructions and drawings. These violations lacked direct safety significance in this instance, but raised concerns about when and how they occurred, who knew, and what records and evaluations exist. This information could help NRC decide whether generic design-control problems existed, which is the remaining thrust of this Unresolved Item.

The licensee conducted an extensive document search and conducted more than 20 interviews with key personnel from construction days. Five records sources at construction, engineering, plant and QA were searched. The retired head of onsite QA was retained to lead the search of the "Construction Vault". Several retirees and employees of non-licensee organizations were among those interviewed. The NRC

has reviewed the nature and results of this investigation. Pertinent information found during this effort was contained in "Containment Meeting Minutes" for Unit 2, and indicated that the construction contractor (Canonie) had installed the hatch covers between May 19 and July 5, 1977, and "fit-up" was required. In retrospect, installation should not have taken so long, nor should "fit-up" have been needed. Canonie was replaced as the site construction contractor in 1985. They were contacted, but they stated they had no pertinent records.

Based on the foregoing, it proved impossible to reach any conclusions from review of this incident, concerning generic design control during construction. The NRC concluded this situation resulted from violation of NRC requirements. However, the matter was identified, reported and corrected by the licensee, it lacked direct safety significance, and measures have been put into effect to prevent this type of violation from recurring. Pursuant to NRC enforcement policy (10 CFR 2, Appendix C) no Notice of Violation is being issued and the inspector had no further questions on the matter.

One violation (Item 2.h - not cited), and no deviations, unresolved or open items were identified.

3. Operational Safety Verification (71707, 71710, 42700)

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, Shift Technical Advisors, and 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.

Evaluation, corrective action, and response for off normal conditions or events, if any, were examined. This included compliance to 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. Status - Unit 1 operated routinely throughout the inspection period, while Unit 2 remained in a scheduled outage.

b. During a control room tour on July 1, 1988, the inspector noted the noise level at the rear of the Unit 1 main control room was rather high due to a loud conversation among several auxiliary equipment

operators at their table. The assigned Unit Supervisor asked the group to quiet down. Initially, he got an uncooperative response. This matter was discussed with the Operations Superintendent, who investigated and intervened promptly, specifically and generically to assure this apparently isolated incident would not recur.

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

4. Radiological Controls (71709)

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. The inspector observed return receipt inspection and surveying for a shipment of anti-contamination clothing coming back onsite after being laundered by an offsite contractor.

b. While conducting an auxiliary building tour with a plant management representative on July 13, 1988, a locally posted status sheet was noted to be in error. The survey sheet posted at the Unit 1 East containment spray pump room was incorrect with respect to the location of the contaminated area within the room and, consequently, the placement of the "step-off pad" for contamination control. While not a requirement, it is the licensee's intent that any changes in radiological status be immediately and accurately reflected on the posted survey status sheet.

c. After turnover of the Unit 2 containment to the Steam Generator Repair Project on June 26, 1988, the inspector experienced some difficulty accessing the Unit 2 containment while complying with the plant-side Radiation Work Permit (RWP) designated for use by NRC - and for licensee management tours. The RWP required notifying plant radiation protection technicians assigned to the containment of the nature and location of planned activities; however, plant technicians were no longer present. Arrangements and RWP adjustments were made to substitute reporting to "project" R-P technicians, who are continuously present.

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

5. Maintenance (62703, 42700)

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 No. 023959: performance of preventive maintenance on CVCS valve 2-QMO-201 per procedure **12 MHP 5030.012.001, "Preventive Maintenance Requirements for Limitorque Motor Operated Valves", Revision 0, dated 2/11/88.

b. Job Order No. 012245: inspect/overhaul of leaking main steam safety valve 2-SV-1B-1. The valve disc had been steam cut and required replacement. This work (and repairs to two other valves suspected of similar leakage) was performed pursuant to procedure **12 MHP 5021.001.005, "Main Steam Safety Valves SV-1, SV-2 and SV-3", Revision 3, dated 1/29/87.

c. Job Order No. 020874: clean and inspect breaker 21D6, the supply breaker for motor control center MCC-2-EZC-D, using procedure **12 MHP 5021.082.002, "ITE Type K600S 600V and 480V Power Circuit Breakers", Revision 4, dated 6/23/88. This Job Order also served as "carrier" for Job Order No. 017772, to replace the solid state trip device on this breaker, and specified device calibration via procedure **12 MHP 5021.082.010, "Maintenance Calibration Procedure for Trip Devices Types SS-13 and SS-14 used on 480V and 600V Power Circuit Breakers", Revision 4, dated 6/23/88.

d. Job Order No. 018113: safety injection check valve 2-SI-158 L4 preventive maintenance and replacement of carbon steel bonnet studs with stainless steel. The stud replacement is in continuation of design change RFC-DC-2718, which will ultimately replace all carbon steel studs which might be exposed to primary coolant (low concentration boric acid) with stainless steel studs. The inspector discussed project status and tracking with licensee representatives. The remaining valves have been divided into four priority categories.

All four categories are scheduled to be completed in Unit 2 during the current outage, though only Category 1 valves are considered mandatory on the basis of current commitments to NRC in response to IE Bulletin 82-02, "Degradation of Threaded Fasteners".

The inspector also reviewed procedure **12 MHP 5021.001.040, "Walworth Aloyco Self Actuating Swing Check Valves", Revision 1 dated 4/17/86. The procedure culminates in a flow verification test which could not be done under existing plant conditions. Because of unusual plant configurations associated with the Unit 2 steam generator replacement, the licensee is accumulating a large number of "incomplete" Job Orders. The licensee has established positive controls to ensure final verification tests will be completed at the end of the outage, when plant conditions permit, and before changing plant MODE.

e. When the inspector learned of problems at another plant involving Limitorque Model SMB-000 valve actuators equipped with melamine switch and cam assembly parts, the licensee was notified and questioned. The subject models were determined to be installed in the auxiliary feedwater injection lines to each steam generator in both Units, and in component cooling water lines providing AFW pump cooling. At D. C. Cook, however, the actuators do not have melamine parts.

f. Documentation packages on the following orders were reviewed:

a. JO. 070759 Adjust 2-WMO-734 to intermediate position as specified by Performance Group.

b. JO. 707960 Adjust 2-WMO-738 to intermediate position as specified by Performance Group.

c. JO. 707961 Adjust 2-WMO-718 to intermediate position as specified by Performance Group.

The above Job Orders were done as corrective maintenance for problems found during surveillance testing.

d. JO. 011227 Replace 2-WMO-736 because valve leaks by.

e. JO. 709507 Replace 2-WMO-738 because valve leaks by.

f. JO. 001859 Replace 1-WMO-754 because valve leaks by.

g. JO. 728022 Replace 1-WMO-731 because valve leaks by.

The Job Orders were performed in mid-1987 and involved installing Centerline brand valves in place of Pratt valves. At the time the valves were replaced, design change controls of the Request for Change (RFC) process were not used, so a 50.59 safety evaluation

was not performed as required. Inspection Report No. 50-315/88004 identified other failures to treat the replacement of valves as an RFC, including a 10 CFR 50.59 review, and issued a Notice of Violation against 10 CFR 50 Appendix B. Since the above changeouts occurred before the violation was identified in Inspection Report No. 50-315/88004, they are considered additional examples of the same violation. A second Notice of Violation is not appropriate. The inspector has confirmed that valve replacements performed since the violation were performed by the Minor Modification Process, which includes a 50.59 review.

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

6. Surveillance (61726, 42700)

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.

a. The inspector examined the essential service water system using System description SD-DCC-HP102; 1-THP SP.128 - "ESC/CCW Flow Balance"; 1-THP 4030 STP.241 - "Unit 1 Essential Service Water System"; 1-OHP 4030 STP.022 - "East & West Essential Service Water System"; Prints OP-1-98416-8 - "Essential Service Water System Elementary Diagrams"; and Print OP-1-5113-23 - "Flow Diagram - Essential Service Water". The observations are as follows:

(1) A system lineup was performed using print OP-1-5113 and 1-OHP 4030 STP.022E and .022W, valve lineup sheet No. 1 to verify that: each accessible flow path valve was in its corrective position; power (visual breakers and fuses) was aligned to actuate on demand; essential instrumentation was operable; and, the valves were numbered and labeled correctly. No problems were identified.

(2) A comparison between 1-OHP 4030 STP.022E and .022W and Technical Specification 3/4.7.4 "Essential Service Water" confirmed that STP.022 properly implements Technical Specification 3/4.7.4

(3) System Description SD-DCC-HP102 (Revision 8 dated May 24, 1988) at Paragraph 3.5.f states in part that essential service water flow to the containment spray heat exchangers is preset at 3300 gpm when in fact it is preset at 2400 gpm. The change

in flow is discussed in a revision to 1-THP SP.128 "ESW/CCW Flow Balance" and is based on a Westinghouse paper NS-SED-1G-17 dated June 11, 1986. This was discussed with the Performance Supervisor and Performance Engineer who agreed to initiate a Change Review Request for the System Description.

(4) Drawing OP-1-5113, Revision 23, "Essential Service Water Flow" lists the essential service water flow to the containment spray heat exchanger as 3300 gpm. This appears to be a design flow that has been revised as discussed above. This was discussed with the Performance Supervisor and Performance Engineer who initiated a Condition Report for this item. The Condition Report is the designated mechanism to initiate a drawing change.

(5) The inspector reviewed the elementary prints for normally closed valves WMO-725/WMO-727 - "ESW to Diesel Generator Heat Exchanger", and WMO-713/WMO-717 - "ESW Outlet valves for Containment Spray Heat Exchanger", and confirmed that the valves will open on selected signals as described in the System Description.

b. Performance of the following tests was observed:

(1) **1-THP 4030 STP.241 - "Unit 1 Essential Service Water System"

(2) **12-THP 6040 PER.001 - "Centrifugal Pump Performance Tests" - performed for return-to-service and baseline data purposes on ESW pump 2 West.

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

7. Fire Protection (71707, 64704)

Fire protection program activities, including fire prevention and other activities associated with maintaining capability for early detection and suppression of postulated fires, were examined. Plant cleanliness, with a focus on control of combustibles and on maintaining continuous ready access to fire fighting equipment and materials, was included in the items evaluated.

a. The licensee submitted required notifications to NRC via his letters of June 21 and July 12, 1988 concerning an extended time (over seven days) fire watches had to be established to compensate for "inoperable" fire barrier walls.

b. Fire protection design problems at other nuclear units, involving control room fire protection, were discussed with the licensee. In one case, a licensee with a "CARDOX" fire suppression system postulated that a suppression line break could saturate the control room with an unbreathable atmosphere without prior warning to the

operators. Another plant with a water fire suppression system postulated a common mode loss of both emergency trains from a fire in one train and subsequent wetting and electrical shorting in the other from the suppression water. Neither of these scenarios proved to be a concern at D. C. Cook.

c. The NRC Licensing Project Manager was accompanied on an inspection tour on July 15, 1988, which focused on areas with installed, automatically actuated, CARDOX fire suppression systems. The licensee is considering a request to NRC to convert some of these areas to manually actuated.

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

8. Security (71881)

Routine facility security measures, including control of access for vehicles, packages and personnel, were observed. Performance of dedicated physical security equipment was verified during inspections in various plant areas. The activities of the professional security force in maintaining facility security protection were occasionally examined or reviewed, and interviews were occasionally conducted with security force members.

a. During auxiliary building tours on June 16 and 17, 1988 the inspector found that a vital area boundary/stairway was being constructed to facilitate passage for the Steam Generator Repair Project (SGRP). The construction of the boundary/stairway was complete except for the installation of the card reader. A guard was stationed to prevent stairway use. The inspector examined the boundary/stairway and directed questions pertaining to traffic flow, construction material and construction design to site security supervisory personnel. Several items appeared deficient. During the discussion, the inspector was informed that the security department had raised similar questions in discussions with Construction. The inspector was told his concerns would be resolved (if applicable) prior to certifying the stairway for use. During subsequent auxiliary building tours the inspector found that his concerns had been resolved.

b. The inspector toured the new steam generator lay down area during non-working hours and noticed a lack of security personnel. This was discussed with senior site management and security controls commensurate with the importance of the components were established.

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

9. Outages (37700, 60710, 61701)

As part of the preparation for the Steam Generator Replace Project (SGRP) the licensee defueled the vessel and stored the fuel in the spent fuel

pool, removed the control rods and drive shafts from the upper internals assembly and stored them in the refueling cavity, and reinstalled the reactor vessel head. The inspector observed:

- Control rod driveshaft removal per Step 2.19 of **2-OHP SP.064
- Reactor vessel head installation per Step 2.20 of **2-OHP SP.064
- Reactor vessel stud tensioning per Step 2.23 of **2-OHP SP.064

The inspector did not identify any problems during these observations.

Following reassembly of the defueled reactor, the following outage milestones were accomplished:

- Walkdown and turnover of containment to the SGRP;
- Installation of the component transport deck and refueling cavity covers;
- Cutting and removal of the concrete upper steam generator enclosures;
- Cutting of main steam and main feedwater piping;
- Girth cutting of steam generators;
- Cutting of primary coolant loop piping;
- Chipping enclosure concrete cuts to expose rebar; and
- Removal of steam dome on one generator.

The inspector observed numerous activities associated with several of the above. One concern was identified. The water in which the control rods are stored has been contaminated by concrete dust, paint chips, and possibly other materials. Concrete dust could be a source of chloride leachate. The licensee expressed a commitment to assuring the control rods are clean before being replaced into the reactor.

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

10. Quality Programs (36700, 39702, 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, and security incident reports. As applicable, corrective action program documents were forwarded to NRC Region III technical specialists for information and possible followup evaluation.

a. During routine review of Condition/Problem reports, the inspector noted an apparent "cluster" of reports dealing with errors made in equipment control ("clearance") process implementation, or outright violations of process requirements. This was discussed with plant management. The Operations Department, which has primary control of the equipment "clearance" process, was developing a series of actions, apparently based on independent identification of the same observation/concern. Disciplinary actions, emphasizing awareness to strict compliance, heightened sensitivity to observing equipment controls already in place, and clarifications to administrative procedures involving how to handle/report discrepancies were all employed. One early result of all the attention to the matter was a reduced threshold and consequent additional "cluster" of reports. This group tended to involve administrative matters (exact tag wording, duplication, sequencing, signoffs) rather than unsafe conditions or loss of equipment status control, and included examples of potential errors found by independent verification (licensee programmatic requirement) before the "clearance" was finalized. The inspector discussed with licensee management how the effectiveness of the effort to enhance this area will be measured. A specific summary report, including evaluation of the numerical frequency of occurrence of events, has been scheduled into the program after six months.

b. Effective June 1, 1988 the following organizational changes were made at the corporate office:

   Mr. R. F. Kroeger, formerly the Quality Assurance Manager, was transferred to the generation engineering division of the Electrical Engineering Department as Assistant Division Manager. In this position he will be involved in engineering management of the D. C. Cook nuclear plant as well as the AEP system fossil and hydro plants.

   Mr. P. A. Barrett was promoted to the position of Director - Quality Assurance, managing the quality program and the Columbus plant site and corporate QA groups.

   Mr. S. J. Brewer assumed the responsibility of Manager - Nuclear Safety and Licensing as well as his current responsibility as Manager - Radiological Support. The dual assignment is temporary.

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

11. 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 to reporting requirements and, as applicable, that immediate corrective action and appropriate action to prevent recurrence had been accomplished.

a. (Closed) Licensee Event Report (LER 315/85025-LL): failure to meet fire watch requirements. When a valve failed during a fire suppression system test, a large number of areas (16) were isolated from fire suppressant. This condition required continuous compensatory firewatch coverage. Due to the number of areas involved, it took 80 minutes (the limit is one hour) to establish all the firewatches. Fire detection capability in the 16 areas was unaffected. The valve failure appeared random and was not predictable from previous valve history. It was replaced and there have been no recurrences of this or a similar event.

b. (Closed) Licensee Event Report (LER 315/85053-LL): potentially inoperable fire suppression. The licensee discovered a damper in the barrier between two fire zones had been designed and installed such that it would automatically close against a fire in one zone only. A fire in the other zone would be less effectively extinguished due to fire suppressant (carbon dioxide gas) loss through the damper. The licensee determined ventilation requirements did not necessitate having an open damper in this wall, so it was permanently closed. There have been no recurrences of this or a similar event.

c. (Closed) Licensee Event Report (LER 315/86003-LL): seismic restraints not previously provided for fire protection piping. The licensee discovered that a 1984 revision to a 1978 design change had not been completed. The original project involved installation of a substantial amount of piping for fire suppression water supply. The revision was issued to ensure that where the added piping was installed above safety related equipment, it would be supported with seismically rated hangers. A total of three hangers remained uninstalled at the time of the LER but a specific engineering evaluation concluded they were not required. Licensee corrective and preventive actions focused on the fact a review was begun in late 1984 concerning the lack of seismic hangers in the original design change, yet the situation was not determined to be adequate until February of 1986.

d. (Closed) Licensee Event Report (LER 315/87014-LL): failure to comply with Technical Specification requirements due to procedural deficiencies. The subject deficiencies were identified by the licensee-sponsored Safety System Functional Inspection (SSFI) and involved three examples of instrument surveillance (response time testing) procedures which did not ensure literal compliance to Technical Specification instructions; certain component actuation

times had not been determined and added into the total for the channel(s) in question. In each case, testing subsequently showed the total response time, including the previously omitted parts, remained well within time limits established in Technical Specifications. The licensee committed to complete review of response time testing procedures and Technical Specification requirements to identify any similar inconsistencies. One additional example was identified and reported as LER 315/88003-LL; see Paragraph 11.h below.

e. (Closed) Licensee Event Report (LER 315/87021-LL): the reactor tripped from 69-percent power at 8:17 a.m. on October 13, 1987, due to a feedwater flow/steam flow mismatch coincident with low steam generator level. The mismatch and low level occurred following loss of the "E" main feedwater pump. The feedwater pump was lost when the alternating current auxiliary oil pump was removed from service (per procedure), because the shaft driven pump failed to maintain lubricating and control oil pressure. Prior to returning the Unit to service the shaft driven pump was inspected and repaired. Inspection of the pump revealed evidence of foreign material damage, however no foreign material was found. Review of post trip data showed that the reactor responded as designed.

f. (Closed) Licensee Event Report (315/88001-LL): the Unit tripped from 90-percent power at 8:21 a.m. on January 13, 1988, because a licensed operator erred during the performance of a reactor trip breaker surveillance test. The operator was requested to close the "B" bypass trip breaker. However, he tried to manipulate (close the already "closed") "B" trip breaker. During this manipulation he depressed the breaker latching mechanism which activated the shunt trip and caused the reactor trip. Review of the trip report and trip data indicated that all systems responded as designed. The inspector reviewed the surveillance test and local labeling and concluded that if the operator had followed the test procedure, this trip was avoidable. The licensee's conclusions were similar. The licensee's corrective actions were administrative in nature and included: counseling and administrative actions with the personnel involved; improvements in pre-job briefings; and, procedure revisions to improve interdepartmental communications and status verification.

g. (Closed) Licensee Event Report (LER 315/87015-LL): the refueling manipulator crane load cell was inadequately calibrated, as discovered in consequence of a crane malfunction. The operations department had performed only a one-point calibration check by weighing the empty refueling mast. Procedures were revised to have maintenance perform a multi-point calibration. Each refueling, operations verifies the calibration has been done as scheduled before using the equipment.

h. (Closed) Licensee Event Report (LER 315/88003-LL): a portion of the power range neutron flux trip circuity associated with the low

setpoint trip (25-percent power) was not response time tested in either Unit 1 or Unit 2. The licensee discovered this discrepancy, involving one bistable, one input relay, and one logic card, during a search for such omissions promised in followup to previous similar problems, especially LER 315/87014. The surveillance procedures are being revised and appropriate testing will be done before a MODE requiring low setpoint trip protection is re-entered.

Items a, d, and h above involved violation of the Technical Specifications. Since these problems were identified, reported and corrected by the licensee, were not repetitive of previous similar problems, had low safety significance, and pursuant to the NRC Enforcement policy (10 CFR, Appendix C), no Notice of Violation was issued for these items.

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

12. NRC Compliance Bulletins, Notices and Generic Letters (71707, 92703)

(Open) NRC Bulletin 88-05.

The licensee's preliminary evaluation identified 56 suspect components, only three of which had been installed in the plant. None were installed in safety-related service. Brinell hardness measurements on the components appeared to all be in acceptable range, though formal evaluation and reporting to NRC remain to be completed.

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

13. Management Interview (30703)

The inspectors met with licensee representatives (denoted in Paragraph 1) on July 27, 1988, to discuss the scope and findings of the inspection. In addition, the inspector asked those in attendance whether they considered any of the items discussed to contain information exempt from disclosure. No items were identified.