U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report Nos. 50-315/88023(DRP); 50-316/88027(DRP)

Docket Nos. 50-315; 50-316 License Nos. 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, MI

Inspection Conducted: September 8 through October 19, 1988

Inspectors: B. L. Jorgensen

J. K. Heller

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

11/8/88
DATE

Inspection Summary

Inspection on September 8 through October 19, 1988 (Report Nos. 50-315/88023(DRP); 50-316/88027(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; emergency preparedness; security; outages; reportable events; Bulletins; and allegations. One Safety Issues Management System (SIMS) item (GI-IE-85-011) (IEB 85-03) was reviewed and remains open.
Results: Of the twelve areas inspected, one violation in one area (Level IV - procedure to control system alignment not followed - Paragraph 2) and no violations were noted in the remaining areas.

The inspection also disclosed weaknesses in the licensee's communication of information on a potentially generic problem to both NRC and other licensees; this despite involvement of safety-related check valves common throughout the industry. See Paragraph 4.a.

The inspection noted strengths in the licensee's approach to Unit operation from a safety standpoint, including a voluntary shutdown to resolve questions about equipment electrical qualification, and performance of both inspection and testing during an unplanned outage. These actions were taken not to satisfy regulatory requirements, but to assure unknown conditions were satisfactory or made satisfactory.

New Unresolved Items were identified (and are discussed in Paragraphs 9.c.) in the following inspection areas: compliance to electrical equipment environmental qualification requirements; and, root cause of inability to transfer concentrated boric acid to borate the reactor coolant system for a shutdown on October 11, 1988.

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, QC/NDE General Supervisor
*D. Krause, I&C/Planning
J. Droste, Maintenance Superintendent
*T. Postlewait, Technical Superintendent - Engineering
L. Matthias, Administrative Superintendent
*M. Horvath, Quality Assurance Supervisor
D. Loope, Radiation Protection Supervisor
*J. Kauffman, Site construction

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 October 20, 1988.

2. Actions on PreviouslyIdentified Items (92701)

(Closed) Unresolved Item (315/88020-01): four of twenty-four hydrogen skimmer system dampers, affecting three volumes, were found to be mispositioned. The licensee has evaluated the "as-found" condition and issued Licensee Event Report LER 315/88008 dated October 6, 1988, which concluded one volume swept by the system could have experienced a post-accident hydrogen accumulation to four percent. This is precisely the limiting design value. The other two volumes involved would have remained below four percent.

The Technical Specifications (T.S.) do not directly address hydrogen skimming system performance and testing standards. The T.S. BASES, however, infer hydrogen control as part of the definition of overall system OPERABILITY. The Safety Analysis Report provides flow values, but not minimum flow values, for the various skimming ducts. The licensee calculated minimum duct flows (to keep hydrogen below four volume percent) in about 1971. These flow values were the basis for flow balance testing during pre-operational testing and they were used in 1985 to find "correct" system damper settings.

Since 1985, the licensee has controlled skimming system configuration (e.g., OPERABILITY) by specifying the "correct" damper settings in Operations procedure 1-OHP 4021.028.008, "Operation of Containment Air

Recirculation and Hydrogen Skimming Systems". The procedure was used prior to entering MODE 4 after an outage. Once set, the dampers were not physically locked in place.

The last damper lineup check via procedure 1-OHP 4021.028.008 occurred in late August, 1987, and is not a dual-verification procedure. Of the four dampers found mispositioned in September 1988, two were checked by one operator and the other two by two different operators. The licensee has been unable to identify a subsequent occasion which would have involved or affected the dampers.

Technical Specification 6.8.1, via reference through Regulatory Guide 1.33 Appendix "A", requires procedures such as that described above be implemented. Failure to control hydrogen skimming system dampers as specified in the procedure is considered a violation of the referenced Technical Specification. Violation (315/88023-01).

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

3. <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.

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. Unit 1 was in routine power operation during the inspection period with the exception of two forced outages and one inspection outage.

   The Unit began the period in a forced outage which extended from September 7, 1988 through September 15. The cause of the outage was increasing reactor coolant system (RCS) leakrate - below the Technical Specification limit. The source of the leak (a 3/8 inch instrument line) was located and repaired, and a number of other inspections and repairs were performed. This included inspection and repair of some Anchor-Darling brand check valves which had internal bolt/stud deficiencies - a situation discussed further in Paragraph 4.a below.

The inspector observed startup activities from the main control room on September 15, 1988, including partial power escalation and rolling off a main feedpump. The licensee's initial approach to criticality early that day had failed to meet the licensee's self-imposed administrative band (within 500 pcm) about the estimated critical position. An evaluation indicated the causes were Boron-10 depletion and the imprecision of the vendor burnup/ Xenon curve. The administrative band was relaxed to 800 pcm and a successful criticality performed. Technical Specifications permit a 1000 pcm band.

b. A second Unit 1 outage occurred on October 11-14, 1988, when inspection and corrective action were needed on pressurizer and reactor vessel head vent system electrical controls - this is discussed further in Paragraph 9.c below.

   The inspector observed startup activities from the main control room on October 14, 1988, including paralleling the main generator and escalating power through the region of manual steam generator level control to placing the automatic controls in service.

c. Unit 2 remained in an extended, scheduled outage throughout the inspection period. Outage milestones and progress are discussed further in Paragraph 9.a below.

d. During a tour of the Unit 2 4KV switchgear room, the inspector found that the CD battery cells were dirty. It appears that dust from construction activities in the adjacent room had been transported via the ventilation system to the battery room. This was reported to a production supervisor who initiated Job Order 029674 to clean the cells.

e. During a tour of the auxiliary building the inspector observed that the sample lines for essential service water from the Unit 1 component cooling water heat exchanger were vibrating freely. Apparently the support braces were not in place. This was identified to the Maintenance Superintendent for corrective action and a review as to why the Unit 1 configuration differed from Unit 2; the latter had support braces in place.

f. Unit 1 tripped from 90-percent power at 3:22 p.m. EDT on October 19, 1988; the last day of the inspection. The cause was determined to be a short in the SSPS Train "B" Logic Cabinet, CRID bus, "power available" light bulb. The bulb failure blew a fuse in the Input Channel I power supply to relays for reactor coolant pump (RCP) No. 1 undervoltage, underfrequency, and power supply breaker position open indication. These false indications of RCP No. 1 failure caused the reactor trip. The inspector responded to the control room and observed initial plant and operator response. Both were satisfactory, with no unusual or unexplained occurrences. Repairs were made, selected testing performed, and the Unit restarted at 2:52 a.m. EDT the following day. The inspector observed the criticality and low level power ascension.

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

4. 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. At the beginning of the inspection period on September 8, 1988, the licensee discovered by physical inspection of the internals that Unit 1 valves 01-SI-151E and 01-SI-151W each had a broken retaining block stud. These inspections had been precipitated by the coincidence of a Unit 1 shutdown (for unrelated reasons) and the discovery of a broken retaining block stud and three cracked studs among ten valves (20 studs) inspected in Unit 2. The Unit 2 inspections were being performed as an adjunct to opening the valves up for replacement of carbon steel body-to-bonnet studs with stainless steel studs; they were "routine" inspections.

   The Unit 1 findings were documented on Problem Report 88-627 dated September 9, 1988. The Problem Assessment Group (PAG) review decided a report to the industry via INPO would be issued and determined the matter was potentially reportable to NRC under 10CFR Part 21 and/or under 10CFR50.73 as a Licensee Event Report (LER). The corporate Nuclear Safety and Licensing group was allocated two weeks for performing a safety and Part 21 evaluation. A like period was authorized the Maintenance and Safety and Assessment groups onsite for determining reportability as an LER. These reviews determined reporting was not mandatory under either Part 21 or LER regulations, however the licensee decided to make voluntary NRC notification. Neither the voluntary NRC nor the industry notifications were expeditiously pursued. The licensee's Unit 1 Monthly Operating Report pursuant to Technical Specification 6.9.1.10, (reported dated October 7, 1988) did briefly discuss identification of and corrective action for the broken block studs found in the subject check valves. This date corresponded with when an LER on the topic would have been due. As a consequence of this routine treatment, NRC became aware of the details of the D. C. Cook findings only after similar findings at the Diablo Canyon Nuclear

- Plant in California surfaced about October 10, 1988 (also not "officially" reported via Part 21/LER) and the valve vendor (Anchor-Darling) advised Diablo Canyon personnel of the D. C. Cook results. NRC concluded the issue had generic potential and issued an Information Notice (No. 88-85) dated October 14, 1988.

The inspector performed a retrospective review of selected licensee actions on this matter and concluded:

i) the initial Unit 2 inspections were not driven by any suspicion of a problem;

ii) a responsible decision was made to inspect in Unit 1 because it was opportune, and another opportunity could be months in coming;

iii) the rationale for selecting which Unit 1 valves to inspect appeared sound; and,

iv) the timeliness of advising other potentially-affected licensee's and the NRC was poor.

These conclusions were discussed at the Management Interview.

b. Job Orders 018414 and 018428: replacement of carbon steel body-to-bonnet studs with stainless steel studs for valves 02-RH-108E and 02-RH-108W respectively. During this design change activity, it was noted the bracket bolts were of two different materials, not appearing to be the specified 410SS-HISI Type 410. The manufacturer (Crane-Aloyco) was contacted and corrective action document generated (Problem Report 88-619). The manufacturer initially reported that, despite appearances, the suspect bolts were 410SS as specified on his drawing. Subsequently, the bolts were identified as ASTM A193 Grade B8, which the vendor confirmed as correct (the drawing was wrong) - so repairs were completed with A193 Grade B8 bolts.

c. Job Order 028354: repair leaking 4-inch weld cap on Unit 2 component cooling water line. Visual inspection showed the weld and a nearby 4-inch branch connection line weld were not full penetration welds as required under ANSI B31.1 piping code. The joint configuration consisted of a square weld prep with fillet weld on the outside diameter only. Problem Report 88-629 was generated to track investigation and correction of the situation. Actual repairs were completed under the subject Job Order which corrected the condition.

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

5. 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.

The following activities were inspected:

a. Special Procedure **1 THP SP-003, "Power Range High Flux (Low Range) Trip Time Response".

   This test was performed in Unit 1 during the September 7-15 outage in follow up (for that Unit) to LER 315/88003. The LER had described a procedural omission such that the low setpoint power range neutron flux reactor trip circuit had not been response-time tested. The Unit remained at power above 25-percent from discovery of the omission until the September 7 shutdown. In those conditions, the circuits were bypassed. The licensee declared the circuits administratively inoperable and made temporary procedure changes requiring manual reactor trip from 12-percent power on normal shutdown - to assure the untested channels would not be relied on as "operable". This was the shutdown technique used on September 7. Completion of the test allowed restart with "operable" channels and deletion of the temporary manual trip step in the shutdown procedure.

b. Surveillance Procedure **12 THP 4030 STP.207, "Ice Condenser Lower Inlet Doors".

   This test was also performed during the Unit 1 September 7-15, 1988 outage. At the time of the shutdown, an active extension request existed to permit the licensee to exceed the normally-specified testing period. This was because no basis for a reactor outage existed (other than to perform testing) until a refueling outage scheduled after the test interval in Technical Specifications would be exceeded. The unplanned September 1988 outage made otherwise inaccessible areas accessible to test personnel, so the testing was done and the exemption request withdrawn. The test showed one bay of intermediate deck doors had frozen and several flow passages had ice buildup in excess of specifications. These findings were documented and corrected. Further inspection follow up is anticipated in review of a Licensee Event Report on the flow passages.

c. **12 THP SP.122, "Spray Additive Eductor Performance Test" - conducted on Unit 1 West train October 3, 1988.

d. **1 THP 4030 STP.411, "Reactor Trip SSPS Logic and Reactor Trip Breaker Train "B" Surveillance Test (Monthly)".

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

6. Emergency Preparedness (82201)

The licensee declared an Emergency Plan "Unusual Event" at 4:07 a.m. EDT on September 22, 1988 upon notification from the Berrien County Sheriff of a fire and evacuation in progress in nearby Stevensville, Michigan. Unit 1 was in operation at 90-percent power at the time of the event, while Unit 2 remained in an extended outage with all fuel offloaded. Applicable Emergency Plan notifications were made.

The fire developed shortly before 2:00 a.m. EDT at the Plastronics plastic products plant about four miles Northeast of the nuclear plant site. The 20,000 square ft. main plant became totally engulfed, and a nearby 50,000 square ft. warehouse was threatened. An area of about 1.5 square miles North and West of the fire was evacuated as a precaution against fire-generated cyanide or possible other toxic gases.

Site operations, communications and access were not affected. The site supported local emergency officials with near-field meteorological data. When the fire was brought under control and the precautionary evacuation cancelled, the licensee was notified and the "Unusual Event" was terminated at 7:40 a.m. EDT. Followup information from FEMA though the County Sheriff indicated major combustion products to have been carbon dioxide, carbon monoxide, and soot.

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 capabilities 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.

One actual fire occurred during the inspection period, involving the inadvertent ignition of plasticized cloth, cable sheaths and hoses in the Unit 2 lower containment by sparks and slag off the arc-gouging of the steam generator girth weld. The fire brigade organized, dressed and responded, but the fire was extinguished by a locally assigned fire watch within about four minutes - the time the brigade arrived at the containment entrance.

On two occasions, fire door No. 322 (separating the Unit 2 CD diesel room from the ESW pipe tunnel) was found open and restrained by a loop of rymple cloth around the door knob and a nearby conduit. A substantive investigation pursuant to Problem Report 88-650 was initiated to identify and locate the perpetrator. Further review of this matter is anticipated in follow up to the Licensee Event Report planned on the topic.

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

8. 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.

The inspector was contacted by a concerned plant employee during the inspection period. The employee had questions regarding the assignment of responsibility for the radiological contamination monitors in the egress area of the site access control building.

A review of the matter was performed, including Security "Post Orders" and the following procedures:

12THP 6010 RAD.647 "Operation of the Eberline Personnel Contamination Monitor".

12THP 6010 RAD.600 "Personnel Decontamination and Incident Reporting".

PMP 6010 RAD.001 "Radiation Protection Manual".

The inspector concluded existing controls were not being violated, but they created opportunities for confusion by a lack of specificity concerning who is responsible and how their responsibility is exercised. This was discussed with licensee management during the inspection and at the Management Interview.

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

9. Outages (37700, 42700, 86700)

a. The following significant activities involving the Unit 2 steam generator repair project, a number of which were observed by the inspector, occurred during this inspection:

(1) reinstallation and weld out of main steam closure pieces;
(2) complete weld out and heat treat on girth welds;
(3) various weld NDT;
(4) re-bar installation and cadwelding - walls;
(5) install concrete forms - walls;
(6) construct steam generator "doghouse" roof form falsework;
(7) reinstall primary loop insulation;
(8) set and shim lower steam generator lateral restraints;
(9) begin "doghouse" roof re-bar installation; and
(10) begin concrete pours - walls.

The project has lost time, to the schedule, and is now perhaps two weeks behind. This was primarily due to problems achieving satisfactory girth welds and with difficulties emplacing re-bar due to the need to adjust and straighten the stub ends of the remaining embedded bar. The project has continued to undergo periodic specialized inspections by NRC Region III personnel, in several disciplines, which are documented separately.

b. Unit 1 took an unscheduled forced outage from September 7 through 15 to find and correct increasing unidentified reactor coolant system leakage. A pre-established short-term forced outage plan was implemented with appropriate modifications. As noted above, these included inspection of RHR check valves manufactured by Anchor-Darling, and performance of testing not possible during power operation for which a Technical Specification extension was being sought. In addition, inspections of systems and areas not accessible during power operation disclosed a secondary steam leak on a steam generator manway and a few small valve packing leaks. Oil was added to three reactor coolant pumps; these had been inaccessible for about six months during the extended unit run. Also, a CD station battery ground which had been present (sometimes intermittently) for several months, was located by manipulating and/or isolating loads in ways not possible while the Unit was in service. The inspector attended outage planning and status meetings and found the licensee's decisionmaking to be safety oriented.

c. Unit 1 had a brief inspection outage on October 11-14, 1988. A QA audit of Unit 2 electronics associated with the reactor vessel head vent and pressurizer vent control systems had identified questions concerning the environmental qualification (EQ) testing configuration. Junction boxes and flex conduits were found configured such that internal condensation would not drain away, creating a submersion rather than a saturation environment. Upon shutdown and inspection of Unit 1, like conditions were found. These were corrected. Further review of this matter is necessary to determine the regulatory implications - e.g., it appears these conditions may have constituted a violation of E.Q. requirements of 10CFR50.49. Pending this additional review, this is considered an Unresolved Item (315/88023-02).

   During the shutdown, some difficulties were encountered in borating the primary coolant system. Apparently, the normal boration path was plugged (implicating a maintenance activity associated with heat-tracing and insulation) and the "emergency borate" path did not indicate flow when initiated. Boric acid flowpaths specified in Technical Specification 3.1.2.2 include a path from the boric acid tanks, via a boric acid pump and charging pump, to the reactor coolant system. No such path appeared to exist, and the licensee entered a 72-hour ACTION statement. Within about 30 minutes, however, the "emergency borate" path was successfully placed in service. An investigation of the cause and duration of the problem was incomplete at the conclusion of the inspection, so further review of this matter is required. Pending this additional review, this is considered an Unresolved Item (315/88023-03).

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

10. Quality Programs (36700, 37701, 40700)

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 focused 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 Occurrence 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.

The licensee announced a number of corporate organizational changes effective October 1, 1988. Some have impact on D. C. Cook Nuclear Plant, as follows:

a. Dr. J. J. Markowsky was elected to become Senior Vice President and Chief Engineer reporting directly to D. H. Williams, Jr., Senior Vice President-Engineering and Construction. Dr. Markowsky will be responsible for the direction of all engineering functions.

b. A new Nuclear Engineering Department is being formed to provide technical support for the Cook Nuclear Plant. The department will be comprised of engineers and technicians from the Electrical, Mechanical, and Civil Engineering Divisions who are currently working full time on Cook Nuclear Plant activities. Mr. T. O. Argenta has been elected to the position of Assistant Vice President-Nuclear Engineering to manage this new department. Mr. Argenta will report directly to the Senior Vice President and Chief Engineer.

c. The Nuclear Operations Division and the Quality Assurance Division will continue to report to the Senior Executive Vice President-Engineering and Construction.

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) LER 315/87017-LL: Failure to incorporate changes to pressurizer level protection set values into procedures. During

biennial procedure reviews it was discovered that the pressurizer level transmitter spans were incorrectly incorporated in the calibration procedures. However, the setpoint was correctly documented in Engineering Control Procedure (ECP) 12-NI-01 dated January 1977. The licensee determined that the 91-percent administrative limit, when recalculated using the ECP, was really 93.25-percent (Technical Specification limit is 93-percent) and could have been 94-percent when considering instrument drift and setpoint tolerance. No credit is taken for these trips in the accident analysis. The licensee determined that a formal documented control measure did not exist in 1977 to assure that ECPs were incorporated into the calibration procedure. Presently, the applicable ECP is a line item in the "References" section of the procedure. The licensee reviewed a 20-percent sample of ECPs and verified that they were properly incorporated into the calibration procedure. The inspector reviewed the ECP and appropriate calibration procedures and verified that the ECP is properly incorporated.

On September 22, 1988, this LER was submitted to the NRC Region III Enforcement Board. Failure to establish and maintain the pressurizer level setpoint is a Technical Specification violation. However, 10CFR, Part 2, Appendix C at Paragraph V.A. states that a notice of violation will not normally be issued for a violation which meets all the following: identified by the licensee; fits a Severity Level IV or V; was reported, corrected and preventive action taken; and, does not appear to be a violation that could have been prevented by licensee actions for a previous violation. It appears that this LER meets the above criteria; no violation was issued.

b. (Closed) LER 315/87022-LL: Two of three Foxboro pressurizer level transmitters drifted outside of their calibration tolerance. The licensee identified during routine channel calibrations, that two pressurizer level channels had exceeded their Technical Specification Limiting Condition for Operation (LCO) value. Other Foxboro transmitter channels had also exceeded their LCO values from transmitter drift; however, the redundancy criteria were not compromised.

The transmitters were installed during the previous refueling outage as part of an Environmental Qualification upgrade. The licensee indicated it was not unusual for new force balance transmitters to exhibit slightly higher drift during their first calibration cycle following installation. Because the transmitters are primarily mechanical devices with moving parts, they will stabilize following a wear-in period.

The inspector reviewed the licensee's analysis of the event. The calculated channel statistical error allowance was compared to the total channel error allowance assumed in the safety analysis. In all cases, the evaluation showed that none of the transmitter calibration drifts had exceeded the safety analysis.

The licensee performed spot-check calibrations on several of the transmitters following the issuance of LER 315/88022. The inspectors reviewed the calibration data and concluded the transmitters were now exhibiting normal calibration interval drift characteristics. In addition, the Foxboro calibration procedure and training provided to the instrument technicians appeared to be adequate. The licensee has adequately addressed the drift problem and the inspectors have no further concerns on this item

c. (Closed) LER 315/87024-LL: Deficient design results in the failure to provide local shutdown and indication panel fuse/breaker coordination. LER 315/87023 reported a condition that required installation of electrical isolation fuses between various sets of local shutdown and indication panels. During final design review of the installed fuses, the licensee determined that proper coordination did not exist between the newly installed fuses and the upstream breakers. This condition was resolved when a design change directed personnel to install properly coordinated fuses. The subject of this LER is documented in Inspection Report 50-315/88003 and is an example of a Notice of Violation issued in that report. This LER is closed. Additional reviews of breaker/fuse coordination will be documented during close out of the Notice of Violation.

d. (Closed) LER 315/88004-LL: Use of improperly aligned test recorder results in Nuclear Instrument Channel being out-of-specification. While performing routine calibration of power range instrument channel N-41, a time constant was found out-of-specification in the conservative direction. After verifying that the correct procedure was used, the time constant was reset. During supervisory review, the shape of the recorder trace was questioned and a concern expressed that an error had been made. The test was repeated the next day using another recorder. During this test, the time constant was found out-of-specification in the non-conservative direction. The investigation determined that the original recorder had an in-line filter which affected the final reading by approximately 0.5 seconds. When the problem was identified, the time constant was reset. The total time the time constant was improperly set was 17.1 hours. During the 17.1 hours the other three channels were operable. The filter is an integral part of the recorder and is activated by a switch located within the recorder. It appears that the filter was activated during a battery surveillance test. A review of the recorder history log showed that since the filter was believed activated, the recorder had not been used for applications where the filter would have affected the outcome. The licensee has implemented an administrative program to identify, by tag, when the filter is activated.

Technical Specification 3.3.1.1 at Table 3.3-1 lists four operable NI channels and requires that an inoperable channel be placed in trip within one hour and reactor power be reduced to 75-percent or a quadrant power tilt ratio be taken at least once per twelve hours. Failure to comply with Technical Specification 3.3.1.1 is a Technical Specification violation. However, 10CFR Part 2, Appendix C

at Paragraph V.A states that a Notice of Violation will not normally be issued for violations which meet all of the following: identified by the licensee; fits a Severity Level IV or V; was reported, corrected and preventive action taken; and does not appear to be a violation that could have been prevented by licensee action for a previous violation. It appears that this LER meets the above criteria; no violation was issued.

e. (Closed) LER 316/87007-LL: Reactor trip due to undervoltage of the reactor coolant pump busses. The Unit tripped from 80-percent reactor power when a failure of the main generator voltage control system caused an undervoltage dip of the reactor coolant pump busses. The voltage transient was also detected by the safeguard bus undervoltage relays which started one of the emergency diesel generators and initiated load shedding of the "A" Train safeguards busses. The "B" Train bus voltage was approximately 70 volts higher at the start of voltage transient; as such, it did not "see" the voltage dip that would have started the other emergency diesel and the load shedding. All other systems responded as designed. The licensee replaced the power supplies to the automatic manual voltage controllers and replaced a number of SCR modules. After the replacements the Unit was returned to service.

f. (Closed) LER 316/87012-LL: Inadvertent opening of reactor trip breakers caused by personnel error. A reactor trip signal was generated while the Unit was in MODE 3 (reactor trip breakers closed and control rods inserted) when Instrument and Control (I&C) Technicians improperly performed a power range nuclear instrument surveillance. The I&C Technicians were authorized to work on power range N-44 and properly tripped the associated SSPS bistables. However, they began the surveillance on power range N-43. When N-43 was placed in "test", two of four logic was satisfied for power range trips. All systems responded as designed. The licensee investigation concluded that the surveillance procedure and labeling were correct and that personnel error was at fault. The event was discussed with the personnel involved and the I&C Department.

g. (Closed) LER 316/87013-LL: Reactor trip caused by a conservative P-13 permissive setpoint. The reactor tripped from approximately 8-percent power when the turbine tripped from a spurious overspeed trip. Apparently permissive P-13, which could have prevented a reactor trip from turbine trip below 10-percent power, was set too low. This allowed the turbine trip/reactor trip to become unblocked prematurely while power was at approximately 8-percent. The error in the P-13 setpoint was traced back to 1986 when the setpoints were recalculated. It appears an input to the setpoint was not converted from psig to psia. The P-13 setpoint was corrected. In addition, the Unit 1 P-13 setpoint was checked, found set too low, and reset properly. All systems operated as designed.

Two licensee identified violations, for which no Notice of Violation was issued, and no deviations, unresolved or open items were identified.

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

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.

(Open) NRC Bulletin 85-03 and Supplement 1: Motor Operated Valve Common Mode Failures During Plant Transients Due To Improper Switch Settings, SIMS item (GI-IE-85-011).

Item e. of Bulletin 85-03 was assigned to NRR for review. The NRR review was documented in a September 12, 1988 memorandum to E. G. Greenman (NRC-RIII) from C. H. Berlinger (NRC-NRR) and is discussed below:

As requested by action item e. of Bulletin 85-03, "Motor Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings," the licensee identified the selected safety-related valves, the valves' maximum differential pressures and the licensee's program to assure valve operability in their letters dated May 16 and October 10, 1986, and May 28 and November 10, 1987. Review of this response indicated the need for additional information which was contained in the Region III letter dated April 11, 1988.

Review of the licensee's May 11, 1988, response to this request for additional information indicates that the licensee's selection of the applicable safety-related valves to be addressed and the valves' maximum differential pressures meets the requirements of the Bulletin and that the program to assure valve operability requested by action item e. of the Bulletin is now closed.

Item f. of Bulletin 85-03 was also assigned to NRR. The results of the inspections to verify proper implementation of this program and the review of the final response required by action item f. of the Bulletin will be addressed in additional inspection reports. Item f. is still open.

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

13. Allegation (92705)

NRC Region III received information during this inspection period alleging Unit 2 steam generator repair project workers were mishandling their personnel dosimetry in such a fashion that the recorded dose would be falsely low. The inspector made special inspections of work areas to note whether dosimetry was being properly worn - no deficiencies were noted. Additional NRC Region III inspections of this matter will be documented in a future inspection report.

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

14. Unresolved Items

Unresolved Items are matters about which more information is required in order to ascertain whether they are acceptable items, violations, or deviations. Unresolved Items disclosed during the inspection are discussed in Paragraph 9.c.

15. Management Interview (30703)

The inspectors met with licensee representatives (denoted in Paragraph 1) on October 20, 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.

The following items were specifically discussed:

a. untimely reporting to industry and NRC of a potentially generic problem with 410SS retaining block studs in certain Anchor-Darling check valves (Paragraph 4.a);

b. the potential for confusion concerning responsibilities and actions involving the radiological contamination monitors in the departure area of the site access control building (Paragraph 8.);

c. an Unresolved Item involving potential violation of Environmental Qualification requirements for electrical equipment (Paragraph 9.c);

d. an Unresolved Item involving determination of the root cause of failure of both normal and emergency boration paths during Unit 1 shutdown October 11, 1988 (Paragraph 9.c), and

e. the LERs to be closed this inspection were specifically identified (Paragraph 11.).

On October 26, 1988 the Plant Manager was notified concerning the NRC enforcement decision expressed in the Notice of Violation transmitted with and discussed in this report.