

NOTICE OF VIOLATION

Indiana Michigan Power Company

Docket No. 50-315
Docket No. 50-316

As a result of the inspection conducted on November 17 through December 26, 1989, and in accordance with the General Policy and Procedures for NRC Enforcement Actions (10 CFR Part 2, Appendix C), the following violation was identified:

The Unit 2 Operating License (DRP-74, as amended) at Paragraph 2.C(1), authorizes operation of the facility at steady state reactor core power levels not to exceed 3411 megawatts thermal.

Contrary to the above, due to an outdated computer core image tape loaded on October 25, 1989, steady state reactor core power exceeded 3411 megawatts routinely until the error was discovered on November 8, 1989. The maximum power deviation was less than 26 megawatts throughout this interval.

This is a Severity Level IV violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, you are required to submit to this office within thirty days of the date of this Notice a written statement or explanation in reply, including: (1) corrective action taken and the results achieved; (2) corrective action to be taken to avoid further violations; and (3) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

JAN 12 1990

Dated _____

William L. Axelson
for W. L. Axelson, Chief
Reactor Projects Branch 2

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-315/89033(DRP); 50-316/89033(DRP)

Docket Nos. 50-315; 50-316

Licenses No. DPR-58; DPR-74

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

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

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

Inspection Conducted: November 17 through December 26, 1989

Inspectors: B. L. Jorgensen

D. G. Passehl

E. R. Schweibinz

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

1/12/90
DATE

Inspection Summary

Inspection on November 17 through December 26, 1989 (Reports No. 50-315/89033 (DRP); 50-316/89033(DRP))

Areas Inspected: Routine unannounced inspection by the resident inspectors of: plant operations; maintenance; surveillance; security; engineering and technical support; reportable events; allegations; Generic Letters; and, miscellaneous inspection items. The following Safety Issues Management System (SIMS) items were reviewed, with the indicated results: (Closed) Generic Safety Issue GSI 93 and Generic Letter GL-88-03 concerning steam binding of auxiliary feedwater pumps.

Results: Of the eight areas inspected, no violations or deviations were identified in seven areas. One violation was identified (Level IV - licensed steady state reactor core power level exceeded - Paragraph 7.i) in the remaining area. The inspection disclosed weaknesses in the degree of discipline occasionally exercised by licensee employees when conducting important but unsupervised activities. Examples included incorrectly documenting a valve lineup and not using required "in-hand" procedures. Problems were also noted in coordinating activities between different groups, including two examples of groups doing testing not properly coordinated with Operations, and a case of Operations not being provided with valid technical support information.

The inspector noted strengths in the licensee's development of a streamlined equipment control method and in a continued conservative approach to any potential equipment discrepancies.

No new Open Items or Unresolved Items were identified.

DETAILS

1. Persons Contacted

- A. Blind, Plant Manager
- J. Rutkowski, Assistant Plant Manager - Technical Support
- *L. Gibson, Assistant Plant Manager - Projects.
- K. Baker, Assistant Plant Manager - Production
- *B. Svensson, Executive Staff Assistant
- J. Sampson, Operations Superintendent
- E. Morse, QC/NDE General Supervisor
- T. Beilman, Maintenance Superintendent
- J. Droste, Technical Superintendent- Engineering
- T. Postlewait, Design Changes - Superintendent
- L. Matthias, Administrative Superintendent
- *J. Wojcik, Technical Superintendent - Physical Sciences
- *D. Hubble, Quality Assurance Engineer
- D. Loope, Radiation Protection Supervisor
- *M. Barfelz, Senior Performance Engineer

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

*Denotes some of the personnel attending the Management Interview on December 29, 1989.

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

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

Observations of the control room monitors, indicators, and recorders were made to verify the operability of emergency systems, radiation monitoring

systems and nuclear reactor protection systems, as applicable. Reviews of surveillance, equipment condition, and tagout logs were conducted. Proper return to service of selected components was verified.

- a. Both units operated essentially at 100-percent rated thermal power throughout the inspection period except for brief power reductions to restore secondary system chemistry. Unit 1 reactor power was reduced to 57-percent on November 22-23, to clean debris from that unit's feed pump turbine condensers. Unit 2 reactor power was reduced to 57-percent on December 19-20, to repair tube leaks in the West feed pump turbine condenser. Both units ended the inspection period in extended runs in excess of 130 days continuous operation, with Unit 1 approaching its record of 175 consecutive days of continuous operation.
- b. On November 21, the licensee apparently made an inadvertent entry into Unit 2 Technical Specification 3.0.3 as a result of having both trains of Emergency Core Cooling System (ECCS) equipment in a potentially inoperable condition for about five minutes. The NRC was notified pursuant to 10 CFR 50.72.

Plant personnel were performing a Unit 2 pressurizer pressure set surveillance while the West Centrifugal Charging Pump (WCCP) was removed from service for planned work. The East Centrifugal Charging Pump (ECCP) was rendered potentially inoperable when, during the surveillance procedure, an "open" signal was fed to the ECCP's emergency leakoff (ELO) valve. This is discussed further in Paragraph 4.e, below.

Licensee corporate engineers reviewed how well the ECCP would have performed its intended function given the above postulated conditions. The results of their analysis found the ECCP would have performed its function within previously analyzed, acceptable bounds. On that basis, the ECCP was not functionally inoperable and Specification 3.0.3 did not apply.

- c. The computerized Clearance Permit System is fully operational, with approximately 225 standard clearances in the "bank." The plant expects to reap the benefits of the system over the next 17 months as an estimated 3000 standard clearances are entered into the new system (ref. NRC Inspection Report 50-315/89029(DRP); 50-316/89029(DRP); Paragraph 3.e.).
- d. The following Problem Reports were reviewed:

Problem Report (PR) 89-1344: Maintenance performed surveillance testing on the East and West auxiliary building crane per Procedure **12 MHP 4030 STP.015 (Auxiliary Building Crane Interlock Verification) without completing a Plant Manager's Standing Order (PMSO) 113 review. The PMSO was written in response to concerns that the licensee's surveillance test procedures do not contain adequate detail to avoid cross train

problems or prevent noncompliance with Technical Specification. See also Paragraph 4.e below.

Problem Report (PR) 89-1335: During second valve verification for 12 OHP 4030 STP.120 vv (Fire Protection Valve Lineup Verification) it was discovered that 1-FP-504 (fire protection water alarm check valve 7FP-504 inlet shutoff valve) was signed off as being sealed open when in fact the valve was tagged closed.

Both Problem Reports were discussed at the Management Interview.

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

3: 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 (JO) B000736: Replace undersized studs in test orifice flange of Unit 2 East motor-driven auxiliary feedwater pump (MDAFP). Studs of three-quarter inch diameter had been found installed instead of the seven-eighths inch studs specified by approved plant drawings, as discussed in a previous Inspection Report (No. 50-315/89029(DRP); 50-316/89029(DRP)). The previous report incorrectly stated that the undersized studs were immediately replaced. The licensee determined they were adequate for interim service (they had survived over ten years in good condition) and scheduled replacement to coincide with other needed work and with routine testing.
- b. Job Order (JO) B017133: Repack East MDAFP pump shaft seals. This job was coordinated with that noted above.
- c. The licensee has identified selected working procedures as procedures required to be present ("in-hand") at the job site; these procedures are numbered with a leading double asterisk. The inspector did not observe any failures to have "in-hand" procedures when required, but noted Problem Report 89-1214 documented such a



failure. Further, a similar failure was observed during a separate but concurrent NRC inspection by a Maintenance Inspection Team. Failure to use "in-hand" procedures as specified is contrary to the licensee's administrative procedures and has been a rare occurrence. The inspector questioned whether these two cases may be indicative of an adverse trend in this area. This was discussed at the Management Interview.

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

4. 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. **2 OHP 4030 STP.017E, "East Motor Driven Auxiliary Feedwater System Test."

This routinely scheduled test was conducted to fulfill testing frequency requirements but also as a return-to-service test following maintenance on the pump shaft seals as discussed in Paragraph 3.b "Maintenance" above. The test observed by the inspector was not completed due to unsatisfactory performance of newly-installed shaft packing. Other performance parameters appeared acceptable. The inspector verified that a successful test was later completed within the required time frame, following a second repacking.

- b. **1 THP 4030 STP.002, "Reactor Coolant Flow Protection Set II Surveillance Test (Monthly)."

- c. **12 THP 4030 STP.211, "Ice Condenser Surveillance."

The inspector observed a partial test involving ice basket weighing as part of the follow up on an allegation which is discussed further in Paragraph 8.

- d. **2 IHP 4030 STP.119, "Steam Generator 1 and 2 Mismatch Set 1."

- e. On November 21, 1989, the licensee conducted a Unit 2 pressurizer instrumentation test which created a simulated Safety Injection (SI) signal in one train of safety equipment. One affect was to open the emergency leakoff valve (minimum flow protection) for the East Centrifugal Charging Pump (CCP). At the time, the West CCP was

out-of-service for ongoing maintenance. When these activities were reviewed later that same day, it appeared both CCPs may have been concurrently inoperable, because the open emergency leakoff line could "steal" some CCP flow from injection into the reactor in certain accident scenarios.

Based on this apparent cross train involvement, the licensee reported the event to the NRC (the East train was affected for only five minutes) as an inadvertent entry into Technical Specification 3.0.3. Two previous examples of cross train involvement had just been one of the topics of a November 16, 1989 Management Meeting between licensee and NRC Region III officials, and a Notice of Violation for one example was issued with NRC Inspection Report 50-315/89029(DRP); 50-316/89029(DRP).

Subsequent detailed evaluation of the November 21 event showed the East train had remained within analyzed bounds, capable of fulfilling necessary accident safety functions, despite the open emergency leakoff valve. The train was thus legally OPERABLE. Still, the Plant Manager recognized a need to immediately minimize the probability for testing to adversely impact safety equipment without being clearly recognized. This was a common factor in all three cases.

On November 22, 1989, Plant Manager's Standing Order No. 113, "Performance of Surveillance Tests" was issued. This Order requires departmental review, followed by dual, independent SRO-licensed review, of all surveillance test procedures, to assure they properly and clearly denote any affects they may have on equipment or system operability. Such review must precede the next procedure use.

This matter is subject to further review in follow up on the licensee's written response and corrective actions for the cited Violation.

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

5. Engineering and Technical Support

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

The following Problem Reports were of note:

- a. Problem Report (PR) 89-1333: While performing surface preparation inspection it was noted that the mortar between the blocks has deteriorated on both sides of the Auxiliary Building laundry room (block walls). The concern is that there are several safety

related electrical systems and components in contact or near the perimeter of the room which could be adversely affected should a seismic event occur.

Corporate engineering personnel, with assistance from an outside engineering contractor, made a determination on how the problem could affect the plant's ability to shut down safely.

The initial findings were that the walls were capable of carrying the types of loads mandated by Code (National Concrete Masonry Association), and therefore that equipment attached and adjacent to the walls would be safe from adverse affects upon occurrence of a seismic event. Corporate engineering recommended that the mortar joints be tuck pointed (replaced with new mortar). The inspector verified this is being done.

- b. Problem Report (PR) 89-1194: The MODEs 4 and 5 shutdown boron curves do not include allowance for boron dilution accident when on RHR (Residual Heat Removal) cooling. The licensee identified a discrepancy between Unit 2 Technical Specification (TS) Figure 3.1-3 and Unit 2 Technical Data Book (TDB) Figure 4.5. The latter is used to implement the Technical Specification requirement for shutdown margin in MODEs 4 and 5. The TDB figure was nonconservatively based on a constant shutdown margin requirement rather than the curve in the Technical Specification.

Analyses performed by corporate engineers (Nuclear Fuel and Analyses Section (NFA)) found that during the time Unit 2 entered the above MODEs (6/11-6/19/89 and 8/15-8/17/89) there was sufficient boron in the RCS to meet the Technical Specification requirements. Until NFA could perform their analyses, however, a conservative value of 300 ppm was added to the TDB figure should the plant be placed in either MODE. The Unit 2 TDB was revised with the correct curves and issued November 20, 1989. This will preclude any problem for the remainder of the operating cycle.

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

6. Security (71707)

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.

On December 8, 1989, the inspector was informed that an armed contract security officer was suspended when she was found intoxicated in the Owner Controlled Area (OCA) while on patrol. The officer was never inside the protected area on the subject date. A second officer was also

suspended after the licensee's investigation showed that he had observed her crying and acting strangely but took no action. The complete information surrounding this event was forwarded to NRC Region III security specialists for followup.

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

7. 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. (Open) Licensee Event Report LER 315/89002: Failure of main steam safety valves to meet Technical Specification lift setpoint values. Valve testing in March, 1989, found 16 of 20 safety valves lifted a few pounds above or below their specified ranges. All valves were adjusted and retested satisfactorily before return to service. This has been a repetitive problem which the licensee ascribed to a too restrictive allowable lift setpoint range. The inspector discussed two questions with licensee representatives:

- (1) the LER indicates a supplemental report will be submitted by April, 1990, but does not discuss why it is needed - (e.g. what additional information can be provided then?);
- (2) the fact that 13 of the valves were scheduled for overhaul goes unremarked, despite that there must be a connection.

b. (Closed) Licensee Event Report LER 315/89009: Required post-maintenance testing (Type C leak testing) not performed prior to entry into a MODE requiring the equipment to be operable. When one valve was found untested, the unit was shut down and the test performed. A review of over 4,000 other maintenance activities found one additional untested valve, which was also tested. Both valves were leak-tight. The cause of this event was lack of a disciplined interdepartmental system for managing outstanding test requirements. The process was frequently verbal, and allowed deferral of documentation.

The initial problem was discovered on July 5, 1989, and was of significant concern to NRC should it prove to be widespread. The licensee and NRC Region III officials conducted a teleconference on July 6 which led to the licensee's commitment letter of the same date, addressing actions to be taken before unit restart. completion of these actions, including the finding that very few omissions had occurred, led to NRC restart authorization July 7, and the unit was restarted the next day.



In the long term, a formal MODE-related constraint is now emplaced in applicable test tracking documentation. This data is trackable to each involved department.

- c. (Closed) Licensee Event Report LER 315/89011: Incomplete surveillance/retest schedule results in failure to conduct Technical Specification required snubber retesting. Snubber 1-GRC-S519 on the pressurizer spray line failed a lockup velocity test criterion in August 1987. It was repaired, retested satisfactorily and reinstalled, but Technical Specifications also required a retest the subsequent outage (in this case, Spring 1988) and the retest was missed.

The subject snubber was inaccessible when the problem was discovered on September 1, 1989, because the Unit 1 reactor was at power. Historical performance of this and similar snubbers (97-percent functional test passes; zero retest failures) led the licensee to request relief from the Technical Specification requiring the retest in this single instance so the unit would not have to be shut down to perform a test of low probable significance. This relief was granted; the snubber is scheduled for testing the next outage of sufficient duration, not later than March, 1990.

Events b. and c. above involve Violations of regulatory requirements which were identified, reported and corrected by the licensee. They were not repetitive, and they lacked special safety significance. As such, in accordance with the NRC Enforcement Policy (10 CFR 2, Appendix C) no Notice of Violation is being issued on these two items.

- d. (Closed) Licensee Event Report LER 315/89004: Pressurizer safety valve lift point high due to setpoint drift. During an unrelated leakage investigation, one pressurizer safety valve was found to lift 10 lbs. above its specified lift range on the first of four trials. The valve was inspected by a vendor (Crosby Valve Co.) servicemen, who also lapped and polished the disc and seat, and it was returned to service.
- e. (Closed) Licensee Event Report LER 316/89005: Containment Type B and C leak rate exceeds LCO value due to excessive valve leakage. Following an NRC Notice of Violation with Inspection Report No. 50-315/89007(DRS); 50-316/89007(DRS) based on licensee use of the "minimum pathway" method for cumulative Type B and C leakage, the 1988/1989 Unit 2 outage penalty was recalculated using the required "maximum pathway" method. The result was a calculated leakage above $0.60 L_a$ (actual value, $0.61 L_a$). All valves exhibiting significant leakage had already been repaired and retested and "as left" leakage was known to be within limits, so the LER itself was the primary result of the recalculation.
- f. (Closed) Licensee Event Report LER 316/89006: Engineered safety features actuation (reactor trip signal). On February 24, 1989, with Unit 2 in MODE 5, but with the reactor trip breakers closed, at



least two of three steam generator No. 24 level channels apparently sensed a pressure change in the steam generator and tripped low. Several steam/feed flow mismatch channels were already tripped because there was neither steam nor feed flow and the flow transmitter reference legs had not yet all been refilled.

Available automatic safety system responses were all verified to have worked properly. The instrument loops were checked out and closely monitored through startup with no abnormalities noted.

- g. (Closed) Licensee Event Report LER 316/89007: Steam generator low-low level reactor trip signal (MODE 3) during cooldown. One steam generator (No. 21) reached its low-low level setpoint during a combined cooldown and high blowdown evolution for chemical cleanup, because it was initially underfed cold auxiliary feedwater, then the auxiliary feedwater flow increase was too late. Resultant cold shrink reduced level to the 21-percent level trip setpoint before recovering.

Applicable automatic safety functions all worked properly. The operator was counseled concerning communication and coordination of his actions, and any problems being encountered, with other control room personnel.

- h. (Closed) Licensee Event Report LER 316/89013: Calibration during start up caused unexpected safety feature actuation. With Unit 2 in MODE 4 and preparing to start up, a routine instrument surveillance was authorized and underway. This surveillance removed the "block" on the high flux trip signals, so operators verified the reactor trip breakers were open prior to the test. The test was suspended when the intermediate range bistables were found out of specification, and the following shift of instrument technicians converted to a calibration procedure for the bistables. Meanwhile, the reactor trip breakers had been closed for another test involving turbine stop valves. The new group of instrument technicians erroneously believed no trip signals would occur during calibration, so they were authorized and began work. The calibration does involve signals above the trip point however, which (not being blocked) caused the actuation. Available automatic safety functions all operated properly in response to the trip signal. The instrument technician training meetings following the event reviewed the implications relating to incomplete shift turnover and inadequate research to correctly assess the affects of ongoing activities.

- i. (Closed) Licensee Event Report LER 316/89018: Rated thermal power exceeded due to computer constant being changed. The Unit 2 license, as amended, at Paragraph 2.C (1), authorizes operation of the facility at steady state reactor core power levels not in excess of 3411 megawatts thermal. Compliance to this limit is routinely monitored via the Thermal Output Program of the P-250 computer in the main control room. A previous LER (316/89009), as discussed in

NRC Inspection Reports No. 50-315/89018(DRP); 50-316/89018(DRP), was issued concerning a violation of this limit which occurred in Unit 2 in February, 1988, and again in March, 1989. A Unit 1 violation in August, 1988, was also noted. All were a consequence of an unrecognized computer program error of long standing.

As discussed in the referenced Inspection Report, no Notice of Violation was issued for the earlier events because criteria of the NRC Enforcement Policy at 10 CFR 2, Appendix C, were met. This included that the events were not repetitive of previous similar problems. The current event again involved an incorrect program in the P-250 computer, caused by loading an old core image tape still containing the Thermal Output Program errors. An administrative checklist of computer constants (to be verified current upon tape reload) did not include the blowdown constants central to this program error, so it was not recognized that "old" values were being reinserted.

The current event is considered a violation for which a Notice of Violation is appropriate (Violation 50-316/89033-01). This LER will be closed and licensee corrective and preventive actions will be followed up in review of their response to the Notice.

The maximum calculated power while the error was present (October 25 through November 8, 1989) was 100.06 percent. Adding the maximum error of 0.67 percent yields a maximum transient overpower of 100.73 percent. This is less than 26 megawatts above the limit and well within the 2-percent overpower frequently assumed as a starting point in accident analysis. Thus, as in the previous event, the error was not especially safety significant.

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

8. Allegations (92705)

a. (Closed) Allegation (AMS No. RIII-89-A-0051): On April 5, 1989, the Senior Resident Inspector and Resident Inspector met with an individual to discuss the following safety concerns which had been previously discussed with the licensee and were the subject of a February 13, 1989, CATALYTIC letter to the licensees QA Supervisor:

- (1) Electrician expressed concern over the quality of welds on electrical supports.
- (2) Painters expressed concern over training relative to procedures.
- (3) Lack of QC during all phases of coating process.
- (4) Inadequate surface preparation prior to paint application.



(5) QC allegedly inspected weld by touch rather than visual.

The individual felt that the March 6, 1988 response to these quality concerns provided by the QA Supervisor was not adequate. On April 11, 1989, the individual provided the NRC resident office with a list of names (one electrician/welder and three painters) of personnel involved in the concerns.

The electrician and two of the painters were subsequently interviewed relative to their concerns. The inspector also met with the QA Supervisor and his staff. The February 13, 1989, letter referenced above indicated that the site QA Department took detailed notes during a previous meeting. The licensee had addressed the concerns described during this meeting relative to Item 1 in Quality Action Request (QAR) No. 070, 12/22/88, and relative to Items 2, 3, and 4 in Quality Action Request No. 071, 12/30/88. QAR 070 identified, among other things, welds that were undersized, had undercut, and burn through. Review of the licensee's corrective actions indicate that they were appropriate. QAR 071 identified the concerns addressed in Items 2 and 3 as valid, but could not confirm Item 4 (inadequate surface preparation prior to paint application). Adhesion testing was performed for the coatings with acceptable results. Regarding Item 5, that "... while offsite, an ironworker told an I&M supervisor that if "Project" QC personnel could not visually inspect weld, the inspector "felt" the joint to determine acceptability." the following was determined. The name of the I&M supervisor was obtained during the interviews. The inspector discussed this issue with the QA supervisor who was aware of it and produced a December 21, 1988, memorandum discussing it. The memorandum states that the I&M supervisor "was instructed to talk to the ironworker and inform him that if real concerns exist in these areas, he should contact Site Q.A." The inspector talked to the I&M supervisor via telecon and determined the name of the ironworker. The ironworker apparently no longer had a concern after a subsequent conversation in which the I&M supervisor asked him the question "Did the ironworker know if the QC inspector was actually doing an inspection or just checking to see if a scheduled weld, which he could not easily use, had been completed yet?" The ironworker did not know the answer to the question and appeared to have no further concerns.

The inspector had copies of QAR 070 and QAR 071 delivered to the individual involved in the April 5, 1989, meeting with the resident inspectors. Subsequent telephone conversation with him indicated he had reviewed the QARs and the licensee has addressed the concerns, and that if the licensee had provided the QARs or even referenced them in their March 6, 1989, response he would not have needed to come to the NRC with the concerns. This allegation is closed.

b. (Closed) Allegation (AMS No. RIII-89-A-0092): An anonymous allegation was received via telephone calls on June 28, July 7, and July 11, 1989. It was alleged that:

(1) A safety injection valve (2 SI-158) had a substantial bonnet leak (several gallons per minute) and that it continued to leak after it was repaired.

(2) The above valve leaked contaminated water and the allegor may have been unnecessarily exposed.

(4) Some ice baskets were weighed improperly.

Regarding Item 1, the valve was inspected on August 15, 1989, and found to be no longer leaking. This is documented in Paragraph 10.d of Inspection Report No. 50-315/89023; 50-316/89023. Item 2 and the ALARA package for the valve repair was reviewed by a Region III inspector who found the licensee had implemented adequate radiological controls.

Regarding Item 3, a review was performed in an effort to substantiate or refute an anonymous allegation to the effect that ice condenser ice baskets surveillances were not being done properly, and that predetermined weights were recorded rather than actually weighing baskets and recording genuine weights.

The inspection involved two approaches: review of records and observation of activities. The records reviewed were the documented results of weighing 269 ice baskets in January and November, 1989, of which 32 baskets were weighed twice. In no case was the recorded weight for any basket identical for both weighings, but all were comparable.

The inspector selected 24 ice baskets to be weighed under NRC observation. Some had been weighed once previously in 1989, some twice. The licensee received no advance notification on which baskets were to be weighed. During the actual weighing, the licensee's approved procedure was utilized. None of the baskets weighed had identical weights to previously recorded values, but all were relatively close. This indicated the previously recorded values were from valid weighing activities, not invented, or predetermined.

The allegation could not be substantiated nor refuted without a monitored weighing of all 1,944 baskets. The safety implications of the allegation do not merit such an extreme action. The safety function of the ice condenser does not depend on the precise amount of ice in any individual basket, but rather on a generally even distribution of ice among the total. NRC monitors ice condenser surveillance regularly, and has a well justified confidence that the ice condenser is being maintained fully capable of meeting its accident safety requirements, with considerable margin.

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

9. 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.

(Closed) Generic Letter 88-03: Resolution of Generic Safety Issue 93, Steam Binding of Auxiliary Feedwater Pumps. The inspector reviewed the licensees corrective actions to the concerns identified in Inspection Report No. 50-315/89029(DRP); 50-316/89029(DRP) Paragraph 11, and found them to be acceptable. The inspector verified that the licensees review of areas entered by the Turbine Building Auxiliary Equipment Operation (AEO) discussed in Paragraph 3.b of the above mentioned report, and additional AEOs was adequate. The inspector also conducted a review of areas entered by a group of randomly selected AEOs with results consistent with the licensees review. This item is closed.

10. Miscellaneous Inspection Activities (71707)

The inspection included a review of personnel qualifications of individuals receiving new assignments as a consequence of a plant staff reorganization which took effect on November 1, 1989. The qualifications of the new Plant Manager, who took over on October 15, 1989, were also reviewed.

Technical Specifications (No. 6.3.1 for both units) requires members of the facility staff to meet or exceed the qualifications of ANSI N18.1-1971 for comparable positions.

The qualifications of the Plant Manager and of the new Assistant Plant Manager-Production were reviewed against the specifications of ANSI N18.1-1971, Section 4.2.1, "Plant Manager". No discrepancies were noted.

The qualifications of the new Operations Superintendent were reviewed against Section 4.2.2, "Operations Manager". Section 4.2.2 states, in part, ". . . at the time of . . . appointment to the active position the operations manager shall hold a Senior Reactor Operator's License." This was not the case for the incumbent, so this specification was not met. The licensee had previously submitted a Technical Specification amendment request which would take exception to this provision of ANSI N18.1 and specify that the Operations Superintendent "must have held" a D. C. Cook or similar plant Senior Operator License (SOL). NRC had not acted upon this request. The request is similar to a provision of ANSI N18.1-1987 which countenances a previous SOL; NRC has also not endorsed the updated ANSI standard. The new Operations Superintendent previously held a Senior Operator License for D. C. Cook (it expired in mid-1988) so he has demonstrated the knowledge and ability associated with passing the NRC Senior Reactor Operator (SRO) examination. The NRC has previously stated (letter: Varga to Dolan, dated January 26, 1984) that the Technical Specifications and ANSI N18.1-1971 allow the Operations Superintendent not to maintain the SRO license. The lack of an active license "at the time of appointment" is therefore considered a technicality, not a violation of sufficient significance to merit any enforcement or corrective action. Because the new Operations Superintendent has not maintained the SRO license, supplementary requirements (stated in the above - referenced NRC letter) apply. These include having capable and licensed individuals reporting to him who "direct the licensed activities

of licensed operators" as prescribed in 10 CFR 55, and having a position description or other description of duties clearly defining duties and responsibilities as a non-licensed manager. These supplementary requirements were verified to be met.

The qualifications of the new Maintenance Superintendent were reviewed against Section 4.2.3 "Maintenance Manager". No significant discrepancies were noted. In this case, however, a clause stipulates he "should" have nondestructive testing (NDT) familiarity, craft knowledge, and an understanding of electrical, pressure vessel and piping codes. The incumbent has NDT familiarity, but NDT is not a responsibility of the maintenance organization at D. C. Cook. Further, a significant understanding of the various codes is not evident from the individual's employment and training histories.

The qualifications of the new Technical Superintendent - Engineering were reviewed against Section 4.2.4, "Technical Manager". No discrepancies were noted.

This was discussed with plant management at the Management Interview.

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

11. Management Interview (30703)

The inspectors met with licensee representatives (denoted in Paragraph 1) on December 29, 1989, to discuss the scope and findings of the inspection. In addition, the inspector also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspector during the inspection. The licensee did not identify any such documents/processes as proprietary.

The following items were specifically discussed:

- a. examples of employee errors potentially indicating inattentiveness or lack of work discipline (Paragraphs 2.d and 3.c);
- b. examples of miscommunication or miscoordination between groups (Paragraph 2.d, 4.e and 5.b);
- c. the reportable events, with emphasis on the apparent violations (Paragraph 7.b, 7.c and 7.i);
- d. the review of an anonymous allegation (Paragraph 8); and,
- e. the results of reviews of manager qualifications against ANSI N18.1-1971 (Paragraph 10).