U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report Nos. 50-315/92014(DRP); 50-316/92014(DRP)

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

Inspection Conducted: July 1 through August 4, 1992.

Inspectors: J. A. Isom
D. J. Hartland
E. R. Schweibinz
W. D. Pegg

Approved By: B. L. Jorgensen, Chief 8/18/92
Reactor Projects Section 2A Date

Inspection Summary: Inspection from July 1 through August 4, 1992
(Report Nos. 50-315/92014(DRP); 50-316/92014(DRP))

Areas Inspected: Routine unannounced inspection by the resident and region-based inspectors of: plant operations; reactor trips; maintenance and surveillance; reportable events; 10 CFR Part 21 Reports; and, NRC Region III requests.

Results: No violations or deviations were identified in the six areas inspected. One unresolved item was identified relating to failure of safety valves to lift at the proper setpoint when tested.

The inspector noted strengths in the conservative management approach to free the refueling mast from one of the fuel assemblies and in the operation of the Unit 2 turbine at greater than normal running speed during troubleshooting of vibrational problems.

Also, the inspector noted strength in the maintenance department's investigation into the repeated failure of "S" spent fuel pool pump control power fuse.

The inspector noted weakness in the operator knowledge and operation of the startup air ejectors which led to the Unit 2 turbine trip and subsequent reactor trip on July 2, 1992.

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DETAILS

1. <u>Persons Contacted</u>

*A. A. Blind, Plant Manager
*K. R. Baker, Assistant Plant Manager-Production
*L. S. Gibson, Assistant Plant Manager-Projects
J. E. Rutkowski, Assistant Plant Manager-Technical Support
B. A. Svensson, Executive Staff Assistant
*T. P. Beilman, Maintenance Superintendent
P. F. Carteaux, Training Superintendent
D. C. Loope, Radiation Protection Supervisor
L. J. Matthias, Administrative Superintendent.
T. K. Postlewait, Design Changes Superintendent
J. R. Sampson, Operations Superintendent
P. G. Schoepf, Project Engineering Superintendent
G. A. Tollas, Acting Safety & Assessment Superintendent
L. H. Vanginhoven, Site Design Superintendent
G. A. Weber, Plant Engineering Superintendent
J. T. Wojcik,Chemistry Superintendent
S. J. Wolf, Quality Assurance Supervisor

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 August 6, 1992.

2. <u>Plant Operations (71707, 71710, 42700)</u>

The inspector observed routine facility operating activities as conducted in the plant and from the main control rooms. The inspector monitored the performance of licensed Reactor Operators and Senior Reactor Operators, of Shift Technical Advisors, and of Auxiliary Equipment Operators including procedure use and adherence, records and logs, communications, and the degree of professionalism of control room activities.

The inspector reviewed the licensee's evaluation of corrective action and response to off-normal conditions. This included compliance with any reporting requirements.

The inspector noted the following with regard to the operation of Units 1 and 2 during this reporting period:

a. <u>Unit 1 Summary of Operation:</u>

Unit 1 began this inspection period in Mode 5, continuing with the refueling outage that began on June 22, 1992.

The plant entered Mode 6 on July 5, 1992, and core unload commenced on July 9, 1992. On July 10, 1992, the licensee halted refueling operations when the refueling machine grapple assembly did not properly engage the top of a fuel assembly and could not be freed. This problem is discussed further in paragraph 2.c below.

The licensee declared an Unusual Event (UE) at 7:08 p.m. on July 18, 1992, when both diesel generators were declared inoperable after essential service water (ESW) was isolated in response to flooding in the ESW/NESW (non-essential) pipe tunnel. The licensee restored the ESW and terminated the UE at 7:39 p.m. after they determined the source of the flooding to be a ruptured expansion joint on a NESW header.

b. Unit 2 Summary Of Operations:

Unit 2 began this inspection period in Mode 1, at approximately 5 percent power, with continuing attempts to balance the turbine rotor in progress. On July 2, 1992, the reactor tripped from approximately 8 percent power after the main turbine was manually tripped due to loss of main condenser vacuum during troubleshooting of the turbine vibration problem.

The unit subsequently entered Mode 5 on three occasions to support maintenance activities. The first entry into Mode 5 was done to perform a weld repair on a high point vent line on the high head injection pump suction header. The plant was cooled down a second time and reactor coolant system level lowered below the reactor vessel flange level to repair leaks to the #1 and #4 incore thermocouple conoseals and the pressurizer manway. The unit ended the inspection period in Mode 5 for disassembly and inspection of the "C" low pressure turbine and #5 bearing to try to determine the cause of turbine vibrational problems. The licensee was also repairing a recurring leak on the pressurizer manway and several primary plant valve leaks.

c. Fuel Assembly Stuck on Refueling Mast:

The inspector reviewed the licensee's response and corrective actions taken to free a refueling mast from one of the fuel assemblies during a Unit 1 refueling evolution. The inspector noted that the licensee's overall response to this event was conservative and well-managed. Proper procedures and controls were in place to ensure that the evolutions to free the refueling mast from the fuel assembly minimized the chances of damage to the fuel assembly.

On July 10, 1992, at 5:30 p.m., the licensee stopped refueling operations because the refueling mast could not disengage a fuel assembly. The problem was identified through use of the underwater camera to be a gripper arm which had snagged one of the

four hold-down springs. After some unsuccessful attempts to manually free the refueling mast, the licensee decided to disengage the snagged fuel assembly using a hydraulically operated jacking tool. The licensee performed a safety evaluation, held a Plant Nuclear Safety Review Committee (PNSRC) meeting and developed a special operations procedure to disengage the mast from the fuel assembly.

Plant Nuclear Safety Review Committee (PNSRC):

The inspector attended the PNSRC meeting convened to approve the special procedure written for disengaging the mast from the fuel assembly and noted that the licensee's conduct of the meeting was a strength. The Chairman ensured that all process and control questions were satisfactorily addressed and incorporated into a special procedure entitled "Release of Fuel Assembly CC07 From Misaligned Manipulator Crane," **01-OHP SP.099, Revision 0, dated July 14, 1992. Comments recommended for incorporation into the procedure by the PNSRC included a statement to ensure that the mast would be adequately supported by the refueling crane once the mast was freed from the fuel assembly. This prerequisite minimized the possibility of the mast dropping on to the top of the fuel assembly and snagging another section of the fuel assembly once the mast became free. The inspector reviewed the procedure and found it to be comprehensive, clear and well-written.

Observation of the Evolution To Free the Fuel Assembly:

The inspector observed the licensee's activities in containment and in the control room to free the refueling mast from the fuel assembly and noted that the licensee conducted this evolution in a well-controlled and conservative manner. The inspector noted that the licensed senior reactor operator (SRO) in charge of the evolution in containment exercised positive control of contractors performing the freeing evolution. The SRO ensured that the special procedure was followed and steps executed in a controlled and deliberate manner. On July 14, 1992, at about 10:44 p.m., the licensee successfully disengaged the snagged fuel assembly using the hydraulically operated jacking tool which was fabricated for this effort.

Root Cause:

At the conclusion of the inspection period, the licensee had not completed their investigation into the cause of the fuel assembly becoming snagged to the refueling mast. The results of their investigation will be documented in the next resident inspector's report 50-315/92016(DRP);50-316/92016(DRP). The inspector did note that the refueling crane was properly positioned at the correct coordinates for the lift of the fuel assembly located at position "G-11."

d. Observations in Control Room - Raising Turbine Speed to 1930 RPM

The inspector observed the licensee's approach to operating the Unit 2 turbine at speeds up to 1930 rpm to investigate turbine vibration problems. Although operation of the Unit 2 turbine at or near 1930 rpm was unusual, the inspector noted that the licensee conducted the approach to this speed in a well controlled manner and that they experienced no problems as a result of this test. The turbine speed was actually raised to approximately 1906 rpm to observe any potential resonant frequency shift to the normal running speed of 1800 rpm. The test indicated that running the turbine at elevated speed did not appreciably affect the vibration of the turbine.

The inspector also reviewed the licensee's special operational procedure "No-Load Turbine Operation For Identification/Correction of Hi Vibration," 02-OHP SP.100, Revision 0, dated July 28, 1992, and found the procedure to be clear, concise and well written. The inspector observed operators perform the "Turbine Operation At Reduced Vacuum" and "Raising Main Turbine Speed to 1930 rpm" sections and found that the operators were able to perform the procedures as written.

e. Observations in Control Room - Reducing Condenser Vacuum:

Following the July 2, 1992, reactor trip (see paragraph 3 below), the inspector observed subsequent licensee operation of the startup air ejectors to reduce the vacuum in the condensers and found that it was performed well and without the loss of vacuum that led to the reactor trip. The operators used special procedure "No-Load Turbine Operation For Identification/Correction of Hi Vibration," which was issued after the reactor trip to control subsequent vacuum reduction evolutions.

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

3. Reactor Trip (93702)

On July 2, 1992, the reactor tripped from approximately 8% power after the main turbine was manually tripped due to loss of main condenser vacuum. At the time, the licensee was adjusting condenser vacuum from 28.5 to 25.5 in. Hg, using the startup air ejectors, to troubleshoot the turbine vibrational problems. During this evolution, the vacuum decreased at a rate that was greater than expected and the turbine control valves opened automatically to maintain steam flow. This caused turbine first stage pressure to increase above the P-13 setpoint at the time the turbine was tripped. The turbine was manually tripped at 24.8 in. Hg. per the licensee's abnormal loss of vacuum procedure and the automatic reactor trip followed. All safety systems functioned as designed.

During follow-up, the inspectors noted that the licensee did not have a special procedure prepared to control the condenser vacuum reduction evolution. The operators were relying on their operational experience and knowledge of the steam air ejector system . The licensee later issued a procedure which was used during subsequent testing, as discussed in the previous section. The inspectors will evaluate licensee root cause analysis and corrective action in conjunction with review of the Licensee Event Report anticipated on the event.

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

4. Maintenance/Surveillance (62703, 61726, 42700)

The inspector reviewed the following maintenance activity. The focus of the inspection was to assure the maintenance activities were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with Technical Specifications.

Troubleshooting Repetitive MCC Control Power Fuse Melting:

The inspectors reviewed maintenance work packages and associated engineering assessments related to a repeated melting of a 10 amp control fuse for the "S" spent fuel pit pump breaker to determine if an adequate root cause evaluation was performed and if appropriate corrective action was taken. The inspectors determined that a thorough investigation was performed by maintenance personnel and that the cause for the loss of control power was identified. However, long term corrective action to the problem, which was documented in December 1989 for a fan of similar design, is still pending.

The fuse first failed on June 24, 1992. Maintenance personnel subsequently removed the breaker from its cubicle and tested it in the electrical shop. No anomalies were discovered. The electricians installed a new fuse and successfully completed post-maintenance testing of the circuit. The fuse failed again on July 14, 1992. This time, in addition to the functional test of the breaker, the electricians performed an inspection of the cubicle. They discovered some corrosion between secondary terminals and on various leads. They made the necessary repairs to the cubicle and inspected the adjacent cubicles. They discovered water intrusion in the cubicles that house the Unit 2 reciprocating charging pump and the high demand fire pump. The licensee determined that the water was coming from switchgear supply fan 2-HV-SGRS-1A, located above the cubicles. When the fan was running during a hard rainfall, some rainwater was apparently being drawn in through the air intake in the roof. It then leaked out of the ducting and motor housing above the switchgear. Maintenance initiated Problem Report (PR) # 92-1145 to document the water intrusion. The PR also documented that the supply fans create a significant amount of noise and do not adequately cool some of the corner areas of the rooms.

As immediate corrective action, the licensee stopped operation of the fan and will continuously run fan 2-HV-SGRS-4A, which does not have any equipment underneath it, to maintain temperature in the room until a permanent fix is incorporated. Similar actions were also taken in the Unit 1 switchgear room. During initial follow-up interviews with maintenance and engineering personnel and review of supporting documentation, the inspectors noted that the licensee has been aware of the potential for water intrusion since December 1989. The inspectors intend further follow-up on this issue.

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

5. <u>Reportable Events (92700, 92720)</u>

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. <u>(Closed) LER 315/90015-LL: Containment B and C leakage exceeds L.C.O. value due to degradation of isolation valve seating surfaces.</u>

On November 2, 1990, the measured leakage for the Weld Channel Pressurization System Valve Enclosure Manway for 1-ICM-305 (Containment Sump Recirculation Valve) exceeded the L.C.O. value (0.60 La) of Technical specification (TS) 3.6.1.2.b. Also, Containment Isolation Valves 1-CS-442-1 and 1-CS-442-3 for the seal water injection lines to Reactor Coolant Pumps No. 11 and 13 had unquantifiable leak rates.

The excessive leak rate of the valve enclosure manway cover was caused by a portion of the O-ring being out of its channel. The manway cover was last installed on June 10, 1989. It had a leak rate of 1500 sccm when tested on June 11, 1989. The licensee believes that a 12 inch portion of the O-ring was pulled out of its channel when the manway cover was aligned for bolting.

Corrective actions included replacing the O-ring and a retest indicated no leakage. Further, the maintenance procedure for installation of the manyway cover, was to be reviewed by February 28, 1991, to ensure adequate direction for proper fit-up of the manway cover/O-ring seal. The licensee's problem Report 90-1640, which addressed this matter, was reviewed by the inspector with the following observations:

(1) Procedures were reviewed to verify adequate directions existed for manway O-ring installation; none were found to exist. Suggestions for a method to bring the problem to the attention of the workers were made and guidance was requested as to an appropriate course of action on February 19, 1991. However, this issue was never resolved. This appears to be an isolated case, and no other examples were

identified of commitments made in LERs not being completed. The licensee did issue problem report No. 92-1065 to review this issue to determine if additional actions are needed. Also, "Type B and C Leak Rate" Procedure No. ** 1-EHP 4030 STP.203, Revision 0, was issued with a reference to LER-90015 and guidance that, for excessive leakage involving piping penetrations of the weld channel pressurization system, initial investigation should focus on the valve enclosure manways.

(2) The excessive leak rates for the seal water injection lines to reactor coolant pumps No. 11 and 13 were attributed to pieces of neoprene found in the lines. These check valves are downstream of the seal water filters and the neoprene pieces were believed to have come from the seal water filters and either broke off during filter replacement or were trapped by the filter and fell off during removal. This was the first time they experienced such problems. The valves were repaired and the maintenance procedure used for replacement of the Seal Water Injection Filters was reviewed to ensure that adequate steps were in-place to prevent the introduction of material to the system during seal replacement.

The inspector has no further questions regarding this LER and it is considered closed.

b. <u>(Closed) LERs 50-315/90016-LL; 50-316/92006-LL: Failure Of Two Pressurizer Safety Valves To Meet Technical Specification Required Surveillance Test Criteria</u>

These event reports document continuing problems with failure of pressurizer safety valves to lift within the required ranges specified in the Technical Specifications. LER 50-316/89-04 also addressed this problem. The problem is generic, not isolated, and is similar to failures of steam generator safety valves to lift within the required setpoint. LERs 50-315/90016-LL and 50-316/92006-LL are being administratively closed, while the matter of adequate corrective action for repetitive safety valve setpoint drift is considered an Unresolved Item. (Unresolved Item No. 50-315/92014-01).

c. <u>(Closed) LER 316/90007-LL (revision 1): Containment type B and C leakage exceeds LCO value due to degradation of isolation valves seating surfaces.</u>

The cause of the excessive leakage for the containment pressure relief Train-A containment isolation valve was degradation of the seating surfaces. The valve was rebuilt and the gaskets and disc seals replaced.

Problem Report 90-0855 was reviewed by the inspector and the

following observations were made:

Overview of data from previous tests indicated that the leakage had made a large step change in 1989. The leak rates were as follows: 1982 = 0 sccm, 1984 = 60 sccm, 1986 = 0 sccm, 1988 = 35 sccm (as found), 1989 = 1700 sccm (as left). The licensee has a guideline leakage acceptance criteria for this 24 inch valve of 1,440 sccm. This is based on its calculation guideline of 60 sccm per inch of pipe diameter. The ASME Section XI permissible leakage for this valve is 4,200 sccm. Also, a step change criterion (maximum permissible leakage + previous as found leakage)/2, is used to evaluate changes in leakage. In this case (4,200 + 35)/2 = 2,117 sccm, which is still above the leakage found in 1989 of 1,700 sccm. The inspector discussed the use of historical leakrate data with the System Engineer for containment leakrate testing. This data may be used to predict ultimate failure. The System Engineer submitted a comment for the next revision to Procedure **12EHP 403D STP.203 "Type B and C Leak Rate" as follows:

> Add to Section 5.3 Test Variations a paragraph to provide review of all type B and C leak rate data which exceeded guideline leakage acceptance criteria but below ASME Section XI permissible. The data shall be reviewed against historic data, consideration of the composition of seating surface for failure characteristics should be a part of the evaluation and may show a gradual degradation of the component. This upward trend could result in ultimate failure at a later date resulting in loss of containment integrity and more frequent LER's. It should be considered at this time whether the component is repaired or returned to service.
>
> Add to Section 7.0 Acceptance Criteria a review of those Test Volumes with guideline leakage exceeded but below ASME Section XI permissible. The reason for not repairing a component should be documented in the B&C test report and justified as to why the decision was made.

The inspector has no further questions regarding this LER and it is considered closed.

d. <u>(Closed) LER 50-316/90010-LL: Plant Outside of Design Basis Due to Downgrading of Polar Crane Hoists</u>

On October 5, 1990, Whiting Crane Company made a Part 21 notification which derated the polar crane main hoist from 250 tons to 55 tons and the polar crane auxiliary hoists from 35 tons to 8 tons. The Whiting Crane Company found through its design review that there was significant overstress on connection points of the crane trolley. The overstressed connections could result in metal fatigue and crane failure.

The licensee made the required modifications to the main and auxiliary hoists under minor modification 12-MM-133. The Unit 1 modifications were completed on October 11, 1990 and the Unit 2 modifications were completed on September 10, 1990.

e. (Closed) LER 316/90011-LL: Engineered Safety Features (ESF) actuation signal on hi-hi steam generator level, due to personnel error, with the unit shutdown.

The cause of the event was attributed to an operator failing to monitor the steam generator (S/G) levels at a sufficient frequency. Contributing factors were a test procedure which required operation above the S/G high level alert alarm and procedural guidance which required intermittent high auxiliary feedwater (AFW) flow rates for AFW pump cooling.

Corrective actions included isolating the S/G until the level decreased to the desired amount. The level decreased below the hi-hi level setpoint within five minutes of reaching it. Management expectations concerning attention to the control panels were re-emphasized in an October 10, 1990, letter to shift personnel. Problem Report 90-1476 was reviewed by the inspector and the following observations were made:

(1) The reactor coolant system temperature indication cross calibration test procedure required operation with the S/G water level at 55 percent. This is above the high level alert alarm of 49 percent. This procedure was revised to require the S/G water level to be maintained at 33 percent.

(2) The procedural guidance which required intermittent high AFW flowrates for AFW pump cooling was revised to allow operation of a motor driven auxiliary feedwater pump (MDAFP) with the test valve open and consider it operable. This was an interim fix as the emergency leakoff line was originally designed for single pump operation. This has subsequently been changed for Unit 2, and Unit 1 will be modified during the current refueling outage. The return line has been increased in size from one inch to three inches.

The inspector has no further questions regarding this LER and it is considered closed.

f. (Closed) LER 316/90012-LL: Reactor Protection System actuation caused by a steam-to-feedwater flow mismatch in combination with low steam generator level.

The cause of the event was a trip of one of the two Turbine Driven Main Feedwater Pumps (TD-MFWP) due to an erroneous actuation of the turbine thrust bearing wear/shaft position detection alarm and trip device.

Corrective actions included removing the position detection device for inspection, performing a bench calibration on the W-MFWP and E-MFWP position detection devices and proper setup on the MFWP turbine at turning gear and no-load operating speeds. Problem Report 90-1960 was reviewed by the inspector. A new procedure, 2IHP 6030 IMP.240, May 1, 1992, for use by Maintenance was developed from vendor guidelines to assure correct calibration and installation at each refueling outage.

The inspector has no further questions regarding this LER, and it is considered closed.

g. <u>(Closed) LER 316/90013-LL: Reactor Protection System actuation while performing a technical specification (TS) required shutdown due to a decreased plant battery cell voltage.</u>

The cause of the event was an unnecessary actuation of the Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC) due to an improper setpoint. The shutdown that was in progress when the event occurred was due to an overly restrictive condition for battery operability contained in the TS. This condition has been subsequently removed.

Problem Reports 90-1989, 90-2012, and 90-2013 were reviewed by the inspector. Subsequent to this event, additional administrative controls have been adopted to enhance control of setpoints in procedures. PMSO.105 is used to obtain Plant Engineering input, and Maintenance Administrative Process MA6.8-01 "Procedures" September 19, 1990, is used to assure subject matter expert engineer review.

The inspector has no further questions regarding this LER, and it is considered closed.

h. <u>(Closed) LER 315/91001: TS 3.7.9.5 required fire hose station inoperable without required backup fire suppression hose due to inadequate administrative controls.</u>

The cause of the event was inadequate administrative controls to assure fire impairment reviews were based on the current status of the fire protection system.

Problem Report 91-0088 was reviewed by the inspector. A memo was issued on January 18, 1991, to address the lessons learned from this event and defined three interim preventive actions as follows:

(1) Place caution tags on fire hose stations being used as backup fire suppression devices.

(2) When a fire protection impairment requires establishing backup fire suppression capability, the fire hose station numbers being used should be included in the applicable "impairment sheet."

(3) When backup fire suppression is used, the applicable open items sheet for the inoperable equipment should include the fire hose station numbers being used for backup fire suppression.

Long term preventive actions were included in a February 12, 1991, revision to Standing Order No. OSO.071. The inspector has no further questions regarding this LER and it is considered closed.

i. (Closed) LER 315/91002: Emergency diesel generator (EDG) declared inoperable for maintenance without required TS surveillance being completed due to personnel error.

The cause of the event was personnel error. The extra senior reactor operator assigned to supervise the removal of the EDG from service failed to recall the surveillance requirement for an inoperable EDG; and the unit supervisor failed to follow-up with the work activity to ensure the surveillance requirements were met.

Problem Report 91-0189 was reviewed by the inspector. The corrective actions included performing the surveillance immediately upon problem identification. It was successfully completed two hours and five minutes after the EDG was declared inoperable. The lessons learned from this event were reviewed with the involved operators by their shift supervisor. The inspector has no further questions regarding this LER and it is considered closed.

j. (Closed) LER 315/91004: Reactor protection system actuation due to reactor coolant pump undervoltage caused by main generator voltage regulator failure.

The cause of the event was a defective main generator regulation system exciter field current limiter panel. Problem Report 91-0602 was reviewed by the inspector. Corrective actions included changes to operating procedures to require matching of manual and automatic main generator voltage regulator setpoints on an hourly basis, and generator reactive power and field current limitations. The defective equipment has been replaced and will be tested during this outage. The inspector has no further questions regarding this LER and it is considered closed.

k. <u>(Closed) LER 316/91001-LL: A roving firewatch was posted instead of the required continuous firewatch due to misinterpretation of the TS 3.7.9.2 requirements.</u>

Problem Report 91-0173 was reviewed by the inspector. Corrective actions included a February 5, 1991, memorandum issued to Operations Department personnel to address the lessons learned from this event, and the need to consider the turbine driven auxiliary feed pump room and the auxiliary feed pump corridor as two separate areas for application of the TS 3.7.9.2 action statement firewatch requirement. The inspector has no further questions regarding this LER and it is considered closed.

l. <u>(Closed) LER 316/91002-LL: Missed surveillance of a single fire damper due to an inadvertent omission of the damper from the procedure during procedure revision.</u>

Problem Report 91-0294 was reviewed by the inspector. The omission was discovered during a revision to the procedure. The damper was immediately visually and functionally inspected and accepted. No changes to the current administrative controls governing procedure revision review requirements were made or considered necessary. The inspector has no further questions regarding this LER and it is considered closed.

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

6. <u>10 CFR 21 Report (36100)</u>

<u>(Closed) 10 CFR 21 Report 315/90001-PP; 316/90001-PP: A 10 CFR 21 report on the potential for Westinghouse Class 1E thermal/magnetic molded case circuit breaker (MCCB) performance characteristics to deviate from published information.</u>

A verbal report was made on December 7, 1989, followed by a written report on December 12, 1989. The report concerns Westinghouse FB3125L (125-amp) MCCBs, which have been observed to trip before the allowable time band given in the time-current characteristics curve for this breaker.

Problem Report 90-047 was reviewed by the inspector. The report was found to be made in a timely manner and comprehensive in all aspects. The interim corrective actions and the long term procedure changes (**12IHP5030.EMP.006, MCC/VCC Preventive Maintenance and Molded Case Breaker Testing) were also fund to be comprehensive in all aspects.

This item was identified by the licensee and was the basis for a 10 CFR 21 report made by Westinghouse (verbal January 5, 1990 - written January 9, 1990). (Note: the Westinghouse 10 CFR 21 report was closed by the NRC Vendor Inspection Branch in Report No. 99900404.)

The inspector has no further question regarding this issue and it is considered closed.

7. Region III Requests (92705)

The inspector obtained data for emergency diesel generator unavailability during years 1991 and 1992. The information was requested to obtain better information on the current reliability of emergency diesel generators in the industry.

Also, the inspector performed a survey of current licensee practices for retention of objects in the spent fuel pool. The inspector found that the licensee minimized the amount of debris in the pool and recently had removed much of the loose debris out of the pool. The inspector found no objects suspended above the spent fuel area of the spent fuel pool.

8. Unresolved Items

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

9. Management Interview

The inspectors met with licensee representatives as denoted in Paragraph 1 on August 4, 1992, 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 or processes as proprietary.