U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-440/92024(DRP)

Docket No. 50-440 License No. NPF-58

Licensee: Cleveland Electric Illuminating Company
Post Office Box 5000
Cleveland, OH 44101

Facility Name: Perry Nuclear Power Plant

Inspection At: Perry Site, Perry, Ohio

Inspection Conducted: November 21 through December 28, 1992

Inspectors: A. Vegel
E. Duncan
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Approved By: _____ 1/20/93
R. D. Lanksbury, Chief Date
Reactor Projects Section 3B

Inspection Summary

Inspection on November 21 through December 28, 1992 (Report No. 50-440/92024(DRP))

Areas Inspected: Routine unannounced safety inspection by resident and region based inspectors of licensee action on previous inspection findings, licensee event report followup, surveillance observations, maintenance observations, operational safety verification, event followup, engineered safety features system walkdown, and evaluation of licensee self-assessment capabilities.

Results: Of the eight areas inspected, three violations were identified concerning a missed average power range monitor (APRM) surveillance (paragraph 3.d), an inoperable accident monitoring containment pressure instrumentation channel (paragraph 3.e), and failure to maintain adequate cleanliness (paragraph 7.b). In addition, two non-cited violations (NCVs) were identified in the area of licensee event report followup (paragraphs 3.a and 3.b).

The following is a summary of the licensee's performance during this inspection period:

Plant Operations

The reactor plant was operated at or near full power during the inspection period with the exception of a downpower to 80 percent reactor power on December 6, to perform a flux tilt and fix steam leaks. Operator response to two safety relief valves unexpectedly opening on

November 21 was considered good. A violation was cited concerning a personnel error that resulted in an inoperable accident monitoring containment pressure instrument (NOV 50-440/92024-02).

Maintenance/Surveillance

The quality of observed maintenance and surveillance activities was generally good. Repair efforts of the Division 1 diesel generator and residual heat removal pump "B" room cooler were considered good. Poor post-maintenance cleanup practices contributed to poor housekeeping conditions in the plant (NOV 50-440/92024-03). A violation was cited concerning a personnel error that resulted in a missed average power range monitor surveillance test (NOV 50-440/92024-01).

Emergency Preparedness

The licensee conducted on emergency preparedness exercise on December 9, 1992. The results of the NRC evaluation were documented in Inspection Report 50-440/92023.

Engineering and Technical Support

The engineering evaluation and troubleshooting of the residual heat removal loop "B" test return valve was good.

Safety Assessment and Quality Verification

The quality of reviewed event reports was acceptable. The onsite and offsite review committees were evaluated as effective.

DETAILS

1. Persons Contacted

a. Cleveland Electric Illuminating Company

M. Edelman, Executive Vice President - Power Generation, Centerior Energy
R. Stratman, Vice President - Nuclear
#*D. Igyarto, General Manager, Perry Nuclear Power Plant (PNPP)
K. Donovan, Manager, Licensing and Compliance
*M. Gmyrek, Operations Manager, PNPP
S. Kensicki, Director, Perry Nuclear Engineering Department (PNED)
F. Stead, Director, Perry Nuclear Support Department (PNSD)
*H. Hegrat, Compliance Engineer, PNSD
E. Riley, Director, Perry Nuclear Assurance Department (PNAD)
*W. Coleman, Manager, Quality Assurance Section, PNAD
V. Concel, Manager, Technical Section, PNED
D. Conran, Compliance Engineer, PNSD
M. Cohen, Manager, Maintenance Section, PNPP
P. Volza, Manager, Radiation Protection Section
*R. Tadych, Manager, Quality Control Section, PNAD
D. Cobb, Superintendent, Plant Operations, PNPP

b. U. S. Nuclear Regulatory Commission

A. Davis, Regional Administrator, RIII
E. Greenman, Director, Division of Reactor Projects, RIII
H. Miller, Director, Division of Reactor Safety, RIII
J. Hannon, Director, Project Directorate III-3, Office of Nuclear Reactor Regulation (NRR)
L. Greger, Chief, Branch 3, Division of Reactor Projects, RIII
*A. Vegel, Resident Inspector, RIII
E. Duncan, Reactor Engineer, RIII
J. Hopkins, Project Engineer, RIII

* Denotes those attending the exit meeting held on December 28, 1992.

\# Denotes those attending the management meeting held in the Region III office on December 17, 1992

2. Licensee Action on Previous Inspection Findings (92701, 92702)

a. (Closed) Violation (440/91012-01(DRP)): Multiple examples of failure to follow procedures during control rod manipulations on May 17, 1991. During this inspection period the inspectors reviewed licensee corrective actions and assessed effectiveness of

those actions to prevent recurrence. Based on review of licensee documentation, all corrective action commitments were completed. In addition, based on routine observation of control room activities since this event, specifically observation of control rod manipulations during plant startups and shutdowns, operators demonstrated procedural compliance and received proper oversight during control rod movements. The inspectors concluded that licensee corrective actions appeared adequate to prevent recurrence. This item is closed.

b. (Closed) Violation (440/92002-01a and 01b(DRP)): Non-licensed plant operators failed to use existing plant procedures while shifting instrument air afterfilters and while performing an equalizing charge on the Division I station battery. The errors made while trying to perform the evolutions from memory had the potential for a reactor scram.

The inspectors reviewed the applicable licensee documentation and concluded that corrective actions for the violations appeared reasonable and adequate to prevent recurrence of the specific events. However, as noted in the letter transmitting Inspection Report (IR) 50-440/92002(DRP), personnel errors had been a continuing concern at the Perry plant. Since IR 50-440/92002(DRP) was issued, personnel errors have continued to occur. Examples included a reactor scram due to an improperly installed oil gasket on a reactor feed pump turbine and both trains of the standby liquid control system being inadvertently isolated. Both events were documented in IR 50-440/92020(DRP). Although the overall number of events caused by personnel error have decreased somewhat since 1991, continued effort in this area is still warranted. The adequacy of the licensee's efforts to reduce personnel errors will continue to be evaluated in future inspection reports. This item is closed.

No violations or deviations were identified.

3. Licensee Event Report (LER) Followup (90712, 92700)

Through review of records, the following event reports were reviewed to determine if reportability requirements were fulfilled, immediate corrective actions were accomplished in accordance with technical specifications (TS), and corrective action to prevent recurrence had been established:

a. (Closed) LER 50-440/92008-00: On April 15, 1992, control room operators discovered that the "A" residual heat removal outboard containment isolation valve, 1E12-F0027A, had been opened and deenergized for approximately 5 hours without the actions of TS 3.6.4 being taken. At the time of the event, the plant was in operational condition 5 (REFUEL) with core alterations in progress.

Licensee Investigation of Root Cause and Corrective Actions

Root Cause

As discussed in the subject LER, valve 1E12-F0027A was locally opened as part of a tagout restoration. However, the motor control center (MCC) for the valve was not re-energized immediately. With the MCC deenergized the control room did not have remote valve position indication. A control room shift turnover caused a suspension of the tagout restoration activities. When the valve's MCC was re-energized approximately 5 hours later, the control room operators identified that the valve was open and closed the valve. The root cause of the event was an inadequate procedure. The guidance in plant administrative procedure (PAP-1401), "Safety Tagging," was not adequate to ensure the proper "returned condition" for valve 1E12-F0027A.

Corrective Action

As immediate corrective action, valve 1E12-F0027A was closed from the control room. Long term corrective actions were to issue a standing instruction to all licensed and non-licensed operators that required repositioning of MOVs from the control room during tagout restoration. Procedure PAP-1401 was also revised to require repositioning of MOVs from the control room during tagout restoration.

Inspectors Review

The inspectors reviewed the applicable licensee documentation and concluded that corrective actions for the subject LER appeared reasonable and adequate to prevent recurrence. Technical specification 3.6.4 required, in part, that with one or more containment isolation valves inoperable, maintain at least one isolation valve in each affected penetration operable and within 4 hours isolate the affected penetration by the use of at least one deactivated automatic valve secured in the isolated position. Otherwise, suspend all operations involving core alterations. During the approximate 5 hours that valve 1E12-F0027A was open, core alterations were in progress. The licensee's failure to isolate the affected penetration within 4 hours while core alterations were in progress was a violation of TS 3.6.4. This violation was not cited because the licensee's efforts in identifying and correcting the violation met the criteria specified in Section VII.B of the "General Statement of Policy and Procedure for NRC Enforcement Actions," (Enforcement Policy, 10 CFR Part 2, Appendix C (1992)). This item is closed.

b. (Closed) LER 50-440/92010-00: On April 30, 1992, while in operational condition 5, surveillance instruction (SVI) SVI-B21-T1402, "Reactor Water Cleanup Isolation Logic System Functional Test," was commenced as part of an Instrumentation and

Controls (I&C) work order to replace control relays. Specifically, SVI-B21-T1402 was used as part of work order 91-3120 to place the plant in the appropriate condition for relay replacement by inserting a trip signal on the "A" isolation channel. In accordance with the work order, technicians then removed relay 1B21-K0148D for replacement, which unexpectedly tripped the "D" isolation channel. This satisfied the nuclear steam supply shutoff system (NSSSS) logic for a balance of plant (BOP), outboard, containment isolation. Valves which subsequently closed included those in the fuel pool cooling and cleanup system which caused a loss of shutdown cooling. Operators took appropriate actions and restored decay heat removal within 15 minutes. No increase in reactor water temperature was noted.

Licensee Investigation of Root Cause and Corrective Actions

Root Cause:

The licensee determined the root cause for this event was an inadequate work order. Both the I&C planner who drafted the work order and the I&C supervisor who reviewed the work order failed to recognize that the 1B21-K0148D relay was in a channel that would cause a BOP isolation if removed.

Corrective Action

To prevent recurrence, I&C personnel were trained on this event with emphasis placed on the importance of attention to detail in all aspects of work order preparation and review. As part of the established requalification training program, all plant licensed operators were instructed on the lessons learned from this event.

Inspectors Review

The inspectors reviewed applicable licensee documentation and noted that all corrective action commitments were completed. The inspectors concluded that the corrective actions appeared adequate and reasonable to prevent recurrence. Appendix B of 10 CFR Part 50, Criterion V, "Instructions, Procedures, and Drawings," required, in part, that activities affecting quality be prescribed by documented instructions of a type appropriate to the circumstances and be accomplished in accordance with these instructions. Contrary to the above, the licensee failed to ensure that work order 91-3120 was properly written to replace relay 1B21-K0148D. This was a violation of 10 CFR Part 50, Appendix B, Criterion V. This violation was not cited because the licensee's efforts in identifying and correcting the violation met the criteria specified in Section VII.B of the "General Statement of Policy and Procedure for NRC Enforcement Actions," (Enforcement Policy, 10 CFR Part 2, Appendix C (1992)). This item is closed.

c. (Open) LER 50-440/92020-00: On October 23, 1992, an inadequate retest of the lower containment inner air lock door seal pneumatic system pressure switch resulted in a violation of TS requirements.

Licensee Investigation of Root Cause and Corrective Actions

Root Cause

The licensee determined the cause of this event was a drawing discrepancy which resulted in the wrong inner air lock door seal pneumatic system being tested on October 24, 1992. The piping system diagram erroneously identified the pressure switches associated with the upper and lower containment air lock door seal pneumatic systems. A contributing cause of the event was personnel error, inattention to detail. An alternate responsible system engineer prescribed the retest without realizing the impact that the maintenance on the pressure switch had on the integrity of the lower containment inner air lock door seal pneumatic system pressure boundary.

Inspectors Review

Initial investigation of this event was documented in Inspection Report 50-440/92022, dated December 10, 1992. During this inspection period the inspectors reviewed licensee documentation, discussed the event with licensee management and reviewed corrective action plans. On December 18, 1992, the licensee identified an additional drawing discrepancy related to the containment air locks. While troubleshooting the upper air lock inner door to determine the cause for a seal on the door not depressurizing, the system engineer identified that the test connections to the small and large seals were incorrectly identified on the piping system diagram. As a result of this discrepancy, the potential existed to test the incorrect seal. The licensee initiated action to determine if the condition of the seals on all of the doors were indeterminate at any time following initial plant startup. In addition, a walkdown of all the air lock doors was to be conducted to identify any additional drawing discrepancies. Pending the inspectors review of the licensee's investigation results and corrective actions, this LER will remain open.

d. (Closed) LER 50-440/92021-00: On November 1, 1992, during a plant startup, the licensee discovered that the average power range monitor (APRM) gain and channel calibration was not completed as required by TSs within 12 hours after exceeding 25 percent reactor power. On October 31, with the plant in operational condition 1 (POWER OPERATION) at 24.4 percent power, the licensee completed SVI-C51-T0024, "APRM Gain and Channel Calibration," at 8:12 p.m., using feedwater pump inlet feedwater flow data. Subsequently, power was raised to greater than 25 percent at approximately 1:00 a.m. on November 1. At 2:48 a.m. preparations were commenced

to reperform the surveillance but difficulties were encountered with the indicated flow received from the feedwater venturis. Plant procedures required that feedwater flow for the heat balance be obtained from the venturis when greater than 25 percent reactor power. Consequently, SVI-C51-T0024 could not be performed as written. At 3:45 p.m. the oncoming unit supervisor noted that the surveillance had not been performed within 12 hours of reaching 25 percent power. As a result, the provisions of TS 4.0.3 were applied, allowing the surveillance to be completed satisfactorily within the next 24 hours. Following a power increase to 52 percent, SVI-C51-T0024 was completed at 10:30 p.m.

Licensee Investigation of Root Cause and Corrective Actions

Root Cause

The licensee determined the root cause of this event was multiple personnel errors. Inadequate communication, failure to follow procedure, and inattention to detail all contributed to missing this surveillance. Feedwater flow indication problems, which interfered with obtaining a satisfactory power calculation, diverted the operators' attention from the requirement to complete this surveillance within the 12 hour time limit. Proper turnover and proper use of the potential limiting condition for operation (LCO) tracing system could have prevented this event. Contributing to this event were procedural inadequacies.

Corrective Actions

To prevent recurrence the licensee counseled all personnel involved in this event on the importance of proper communication and proper use of the LCO tracking system. In addition, all licensed operators were to review this event as part of requalification training. The operating and surveillance instructions involved in this event were to be revised to include specific time limitations for completion of the APRM calibration.

Inspectors Review

As documented in IR 50-440/92022(DRP), the inspectors previously evaluated the event and the licensee's immediate corrective actions. During this inspection period, the inspectors reviewed licensee documentation of the event, including investigation results and long term corrective actions. As noted in paragraph 2.b., the issue of personnel errors is still of concern. The licensee has implemented a program to trend and reduce personnel errors; however, the corrective actions for previous personnel errors have apparently not been fully effective. Technical specification Table 4.3.1.1-1, footnote (d), required that APRM channels be calibrated to conform to the power values calculated through heat balance during operational condition 1 when thermal power was greater than or equal to 25 percent power.

[illegible] y, the provisions of TS 4.0.4 were not applicable, provided the surveillance was performed within 12 hours of reaching 25 percent power. Technical specification 4.0.4 prevented entry into an operational condition unless the surveillance requirements associated with the LCO had been performed within the applicable surveillance interval. Contrary to the above, on November 1, 1992, after reaching 25 percent reactor power at 1:00 a.m., the APRM gain and channel calibration was not completed within 12 hours which resulted in a violation of TS 4.0.4 and TS Table 4.3.1.1-1, footnote (d). This is a violation (50-440/92024-01(DRP)).

e. (Closed) LER 50-440/92023-00: On November 8, 1992, while in operational condition 1, control room operators identified a discrepancy between the two wide range containment pressure channels displayed on the Emergency Response Information System (ERIS). (This event was previously discussed in IR 50-440/92022(DRP)). On November 10 the licensee determined that containment pressure transmitter D23-N270A was inoperable due to an improper valve lineup. The pressure transmitter and instrument loop for D23-N270A were satisfactorily calibrated and D23-N270A was returned to service on November 11. Containment pressure transmitter D23-N270A was inoperable from March 26, 1992, until November 11 without the actions of TS 3.3.7.5, Table 1, being taken.

Licensee Investigation of Root Cause and Corrective Actions

Root Cause:

The licensee determined that the root causes of this event were an inadequate SVI and inattention to detail on the part of the I&C technicians who returned D23-N270A to service. The steps in SVI-D23-T2002, "Containment Atmosphere Monitoring Isolation Valves Seat Leakage and Position Indication Test," that isolated and later restored pressure transmitter D23-N270A to service were inadequate because they did not specify the pressure transmitter sensing line isolation valve. Those steps were not in conformance with PAP-0517, "Preparation of Technical Specification Surveillance Instructions," which required that SVIs "should be written to stand alone." The I&C technicians that restored D23-N270A to service had additional drawings in the field that identified the correct isolation valve and the drawings and valves were properly labeled. Nevertheless, the I&C technician opened the instrument manifold test valve instead of the instrument sensing line manifold isolation valve, which resulted in the instrument remaining isolated.

Corrective Actions

To prevent recurrence, the licensee initiated action to revise the surveillance instruction to explicitly identify applicable

instrument valves and develop procedural guidance to ensure applicable instrumentation is in service prior to operation condition changes. In addition, the technician who restored the transmitter to service on March 26, 1992, was counseled and lessons learned from this event were to be reviewed by all I&C technicians, supervisors, and surveillance writers.

Inspectors Review

In IR 50-440/92022(DRP), the inspectors documented initial event occurrence. During this inspection period, the inspectors reviewed licensee documentation of the event, specifically the investigation results and corrective actions. In addition, the inspectors reviewed drawings and procedures and conducted a walkdown of the affected instrumentation. The inspectors concluded that the licensee's investigation appeared thorough in reviewing the event and adequate in attributing the cause to a combination of inadequate procedures and inattention to detail. As noted previously, the corrective actions for previous personnel errors have not been fully effective.

Technical specification Table 3.3.7.5-1, Action 80-a, required, in part, that with the number of operable accident monitoring containment pressure instrumentation channels less than two, restore the inoperable channel to operable status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Contrary to the above, from June 9, 1992 until November 10, 1992, with the exception of some short periods in which the plant was shut down for repairs, the plant was in operational conditions 1, 2, or 3 with less than the two required channels of accident monitoring containment pressure instrumentation, without the required actions of TS Table 3.3.7.5-1, Action 80-a, being taken. This is a Violation (50-440/92024-02(DRP)).

No deviations were identified; however, two violations and two non-cited violations (NCVs) were identified.

4. Monthly Surveillance Observations (61726)

For the surveillance activities listed below, the inspectors verified one or more of the following: testing was performed in accordance with procedures; test instrumentation was calibrated; limiting conditions for operation were met; removal and restoration of the affected components were properly accomplished; test results conformed with technical specifications, procedure requirements, and were reviewed by personnel other than the individual directing the test; and any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

Surveillance Activity	Title
SVI-C11-T0223A	Setpoint Channel A Calibration for 1C11-N054A.
SVI-E22-T1319	Diesel Generator Start and Load Division 3

No violations or deviations were identified.

5. Monthly Maintenance Observation (62703)

Station maintenance activities of safety-related systems and components listed below were observed and/or reviewed to ascertain that activities 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 operations 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 were inspected as applicable, functional testing and/or calibrations were performed prior to returning components or systems to service, quality control records were maintained, activities were accomplished by qualified personnel, parts and materials used were properly certified, radiological controls were implemented, and fire prevention controls were implemented.

Work requests were reviewed to determine status of outstanding jobs and to assure that priority was assigned to safety-related equipment maintenance which may affect system performance.

During the inspection period, the inspectors noted two examples of maintenance activities that were performed well. On December 1, 1992, the licensee replaced the Division 1 diesel generator left bank number 1 cylinder gasket following identification of a jacket water leak. The maintenance activity was planned for approximately 60 hours but was completed in approximately 40 hours. In planning for this repair, the licensee contacted other plants for information on repairing the leak, and used a Unit 2 diesel generator as a mockup for training the maintenance crews. As a result of incorporating lessons learned from other plants and the use of the mockup, the time that the diesel generator was inoperable was minimized. The second example involved the repair of the residual heat removal "B" (RHR-B) room cooler on December 5-16, 1992. On December 5, the RHR-B room cooler failed due to a fan belt coming off and failure of the fan bearing and housing. The room cooler was subsequently repaired on December 6, by replacing the damaged components with parts obtained from Unit 2. Due to the prompt corrective maintenance effort, the licensee minimized the time that the RHR-B loop was not operable.

Specific Maintenance Activities Observed:

Work Order (WO)/Repetitive Task No.	Title
WO-920005064	Division II Diesel Generator Fuel Injector Sticks
WO-R85-9795	Control Room HVAC Return Fan "B"
WO-910004014	Replace Unit-1, Division-2 Battery
WO-R85-00286	1P450130A Limitorque Operator Maintenance per PMI-0030
WO-92-03641	Install Lube Oil Reservoir Drain Valve, Condensate Booster Pump C

No violations or deviations were identified.

6. Engineered Safety Features System Walkdowns (71710)

In addition to routine observations made during regular plant tours, the inspectors conducted walkdowns of the accessible portions of selected safety-related systems. During this inspection period the inspectors conducted a walkdown of the high pressure core spray (HPCS) system. The inspectors verified system operability through reviews of valve lineups, system prints, equipment conditions, and control room indications.

As a result of the walkdowns, the inspectors noted that the general condition of the HPCS system was good. The system was aligned in accordance with the appropriate valve lineup sheet, the HPCS pump and valve rooms were well lit, and the components were properly labeled. In the HPCS pump room, several general housekeeping deficiencies were identified:

- Debris on floor under grating.
- Debris under the HPCS waterleg pump skid.
- Loose parts on the HPCS waterleg pump skid.

There was a small packing leak on HPCS condensate storage tank suction valve, 1E22-F001. A catch basin was under the valve to direct any leakage to a drain and a WO had been previously written to repair the leak. Additionally, there was a slight amount of oil dripping from the manual actuator of the HPCS pump discharge check valve bypass valve, 1E22-F026. A maintenance work request was written to investigate and repair the leak.

In the HPCS valve room, several general housekeeping deficiencies were identified:

- Several tools laying on piping insulation and on pipe supports.

- Debris on floor including rags, plastic tie-wraps, and tape.

- Drain hoses laying on the floor.

The inspectors noted that the deficiencies were of relatively little safety significance. However, they were indications of inadequate post-maintenance cleanup, inattention to detail, and overall poor housekeeping practices. Upon being notified of the deficiencies by the inspectors, the licensee took action to correct the deficiencies. The general material condition of the plant is further discussed in paragraph 7.b. of this report. The inspectors concluded that the observed condition of the HPCS system appeared adequate to support performance of the systems intended safety function.

No violations or deviations were identified.

7. Operational Safety Verification (71707)

The inspectors observed control room operations, reviewed applicable logs, and conducted discussions with control room operators during this inspection period. The inspectors verified the operability of selected emergency systems, reviewed tagout records, and verified tracking of limiting conditions for operation associated with affected components. Tours of the pump houses, control complex, the intermediate, auxiliary, reactor, radwaste, and turbine buildings were conducted to observe plant equipment conditions including potential fire hazards, fluid leaks, and excessive vibrations, and to verify that maintenance requests had been initiated for certain pieces of equipment in need of maintenance. The inspectors by observation and direct interview verified that the physical security plan was being implemented in accordance with the station security plan.

The inspectors observed plant housekeeping, general plant cleanliness conditions, and verified implementation of radiation protection controls. In addition, the inspectors observed construction of the low level radioactive waste building.

a. Reactor Fuel Leak

During the course of the current operating cycle, an increase in coolant activity had been noted due to leaking reactor fuel. In September 1992, the dose equivalent iodine (I-131) concentration was .005 microcuries per gram. At the end of the inspection period activity levels measured were approximately .012 microcuries per gram. Technical specification 3.4.5a requires specific activity of the primary coolant to be limited to less than or equal to 0.2 microcuries per gram dose equivalent I-131.

As the coolant activity increased during the operating cycle, the licensee increased sampling frequency and performed two flux tilts to localize the location of the leaking fuel. The flux tilt involved the sequential insertion and withdrawal of control rods in various locations of the reactor core while monitoring for changes in offgas system pretreat activity. As a result of the flux tilts, the fuel leak was localized to one fuel bundle. To suppress the release rate, control rods near the leaking bundle were inserted. Coolant activity levels initially decreased following the control rod insertions, but eventually continued to increase. Concurrent with increased coolant activity, plant dose rates also increased which impacted personnel accessibility to the plant. As a result of the effect on the plant by the fuel bundle leaks, the licensee announced on November 30, 1992, plans to perform a mid-cycle outage commencing January 8, 1993. During the mid-cycle outage, the leaking fuel bundle were to be removed.

The inspectors evaluated licensee actions in response to the fuel leak, including the licensee's monitoring of plant areas for changing radiological conditions due to increased coolant activity. The inspectors concluded that licensee efforts appear to be conservative in minimizing the effects of the fuel leak on offsite and onsite radiological activity levels.

b. Plant Material Conditions

Based on inspectors observations of plant areas, a decline in the general housekeeping and condition of plant equipment was noted during this inspection period. Specific deficiencies noted include:

Plant Area		Discrepancies
Reactor Core Isolation Pump Room:	-	Debris below floor grating: tools, scaffolding material, plastic bags, face shield, insulation material laying around loose.
Heater Bay:	-	Reactor feed pump turbines (RFPT) A and B have numerous oil leaks of approximately 0.5 gals./day.
	-	Insulation not installed on B RFPT contribute to high room temperature.
	-	RFPT B has tools laying around loose.

Low Pressure Core Spray Pump Room, "C" Residual Heat Removal pump room, and High Pressure Core Spray Pump and Valve Rooms:	-	Tools, hoses, plastic bags laying around loose below grating.

The above deficiencies indicate continued poor post maintenance cleanup practices and inattention to detail. The inspectors discussed these deficiencies with licensee management and action was taken to correct the problems. The inspectors previously documented the status of the housekeeping and material condition of the plant in IR 50-440/92002(DRP) dated March 13, 1992, 50-440/92012(DRP) dated July 22, 1992, and 50-440/92022(DRP) dated December 10, 1992. Though the decline in housekeeping by itself was not of safety significance, it was an indicator of a lack of attention to detail and poor maintenance practices. In addition, the impact poor cleanliness practices had on plant operations was demonstrated during the plant restart from the third refueling outage. As previously documented in IR 50-440/92012(DRP), several plant transients resulted from post maintenance debris contributing to the clogging of the hotwell pump suction strainers. Appendix B of 10 CFR Part 50, Criterion II, Quality Assurance Program required, in part, that activities affecting quality shall be accomplished under suitable controlled conditions. Controlled conditions include the use of appropriate equipment, suitable environmental conditions for accomplishing the activity, such as adequate cleanliness, and assurance that all prerequisites for the given activities have been satisfied. Contrary to the above, based on the inspectors observation of debris below the grating in the reactor core isolation cooling pump room and observation of oil leaks and debris in the reactor feed pump turbine rooms, the licensee failed to maintain adequate cleanliness. This is a violation (50-440/92024-03(DRP)).

c. Training Observations

During the report period, the inspectors attended the licensee's general employee training and radiological controls training (RCT). For the training observed, the inspectors noted that pertinent course material was available to each trainee and classroom lectures were provided by knowledgeable licensee personnel. Of note was the practical exercise required for successful completion of RCT training which included proper donning and removal of protective clothing. Based on the observations noted above, the inspectors concluded that the training provided was well planned and useful for the attendees.

d. Residual Heat Removal "B" Test Valve Failure

On December 22, 1992, at 6:29 a.m., while securing from the suppression pool cooling mode on the RHR-B system, valve E12-F024B, RHR-B Test Valve to suppression pool, lost power while

being stroked closed. Subsequent investigation determined that two mainline fuses to the valve motor were blown. The valve is an 18 inch motor operated gate valve, with three 12 amp mainline fuses installed to protect the valve motor.

The licensee entered the TS 3.6.3.3 LCO action requirements for the "B" suppression pool cooling loop being inoperable. The LCO requires restoration of the loop to operable status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours. The licensee initiated troubleshooting of the valve to determine the cause for the failure. Various possible causes were reviewed including:

- valve wedge being cocked
- crud in seat of valve
- motor electrical fault
- inadequate stem lubrication
- bent stem
- limit switch degradation/setting
- guide rib wear/high spots/nicks
- fuses degraded
- valve mechanism degradation

As a result of evaluating the above possible causes, the licensee concluded that the most likely contributors to the event were valve degradation due to wear, coupled with some fuse degradation over time with valve usage. Based on successful stroking of the valve during troubleshooting and assessment of degradation of the valve, the licensee evaluated the valve as operable on December 24, 1992, at 9:52 a.m. To minimize wear, administrative controls were placed on the usage of the valve until the January 1993 maintenance outage. During the outage, testing will be performed on the valve to further evaluate the physical condition of the valve. In addition, the licensee initiated plans to evaluate the possibility of replacing the valve with a globe valve during the outage.

The inspectors, including NRC regional and headquarters staff personnel, reviewed the licensee's investigation of the valve failure and discussed the results with the plant engineering

staff. Based on that review, the inspectors concluded the licensee's efforts appeared thorough and reasonable and in accordance with TS.

One violation concerning inadequate cleanliness was identified; no deviations were identified.

8. Onsite followup of Events at Operating Power Reactors (93702)

a. General

The inspectors performed onsite followup activities for events which occurred during the inspection period. Followup inspection included one or more of the following: reviews of operating logs, procedures, and condition reports; direct observation of licensee actions; and interviews of licensee personnel. For each event, the inspectors reviewed one or more of the following: the sequence of actions, the functioning of safety systems required by plant conditions, licensee actions to verify consistency with plant procedures and license conditions, and verification of the nature of the event. Additionally, in some cases, the inspectors verified that the licensee's investigation identified root causes of equipment malfunctions and/or personnel errors and the licensee was taking or had taken appropriate corrective actions. Details of the events and licensee corrective actions noted during the inspector's followup are provided below.

b. Details

(1) Unexpected Safety Relief Valve Actuation

On November 21, 1992, at 3:57 a.m., with the reactor operating at 99 percent power, two safety relief valves (SRVs) opened for approximately 1 minute resulting in a plant transient.

On November 21, at approximately 12:00 a.m., during a panel walkdown, a plant operator identified that SRV Reactor Pressure Low Low Set trip units 1B21-N0617B and 1B21-N0618B were in the tripped condition. The tripped units were declared inoperable and an investigation was initiated to determine the cause. While performing a step in surveillance procedure SVI-B21-T0369-B, "SRV Pressure Actuation Channel B Functional for 1B21-N668B," as part of the troubleshooting effort, the two SRVs opened. Specifically, as the technician pulled the Calibration Select/Command switch on the calibration unit to verify that it was pulled out in accordance with the surveillance procedure, SRVs 1B21-F0051C and 1B21-F0051D opened. Plant operators responded in accordance with Off Normal Instruction ONI-B21-1, "SRV Inadvertent Opening/Stuck Open," and attempts were made to close the valves. Upon resetting

the "B" Low Low Set Logic, the SRVs closed. The SRVs were opened for approximately 1 minute and 12 seconds, with no change in containment parameters being noted. Upon SRV closing, reactor pressure increased slightly resulting in a power spike to 102.6 percent reactor power. At 3:58 a.m. power was reduced due to the power spike exceeding rated thermal power. Following engineering evaluation of the effect on the reactor vessel by the two SRV actuations and determination that no damage occurred, reactor power was restored to 99 percent at 4:59 a.m. At approximately 7:02 a.m. the licensee informed the NRC operations center of this event via the Emergency Notification System (ENS).

The licensee's initial troubleshooting efforts were directed at recreation of conditions present at the time the event occurred. During this troubleshooting the problem was not duplicated. Subsequent inspection of the power supply, individual trip modules, circuit cards, and instrument racks also did not identify any potential problems that could have caused the event. The licensee replaced trip units 1B21-N617B and 1B21-N618B and the calibration unit, though no specific problems were identified with these units. The licensee preliminarily concluded that the likely cause of the SRV actuation was an unidentified electrical perturbation coupled with the effected trip units being apparently more susceptible to noise due to possible circuit component degradation over time. The licensee's investigation of the causes for this event was continuing. The trip unit vendor was involved with the troubleshooting efforts.

The inspectors reviewed plant and operators' response to the SRV actuation and monitored licensee troubleshooting efforts. The inspectors concluded that the operators responded in accordance with plant procedures and plant response was as expected. Licensee initial investigation efforts appeared thorough.

The licensee initiated condition report (CR) 92-271 to document the results of their investigation into the cause of this event and corrective actions taken. In addition, LER 50-440/92024 was submitted on December 21, 1992 in accordance with 10 CFR 50.73. The inspectors will review that report in a future inspection period.

(2) Oil Spill

On December 8, 1992, at 2:50 p.m. the licensee discovered an oil leak in the "A" main lube oil heat exchanger to the service water system. Upon review of lube oil usage data, approximately 700 gallons of main turbine lubricating oil was unaccounted for and had possibly been discharged to Lake

Erie via the service water system over an 11 week period. Subsequently, the "A" heat exchanger was sampled for oil intrusion with no leaks identified. The licensee initiated CR-92-279 to document the event and track corrective actions. The licensee notified local and state authorities of the potential oil release in accordance with PAP-0806, "Oil/Chemical Release Contingency Plan." On December 8, 1992 at 3:22 p.m. the licensee informed the NRC Operations Center of the event via the ENS.

No violations or deviations were identified.

9. Evaluation of Licensee Self-Assessment Capability(40500)

a. On-Site Review Committee

During the report period, the inspectors observed an on-site review committee meeting to evaluate that organizations effectiveness. For the meeting attended, the inspectors considered the following attributes: the degree of plant management involvement and/or domination of discussions; if constructive discussion occurred; if the majority of the committee consistently voted the same as the chairperson; if the committee was biased toward operation or safety; and, if the committee used design basis, the Updated Safety Analysis Report, or vendor technical manuals for their determinations in addition to the TS.

In preparation for the meeting, the inspectors reviewed the draft submittals given to the on-site review committee for approval. Items presented to the on-site review committee included safety evaluations, temporary changes to procedures, setpoint change requests, procedural revisions, and design change packages.

During this report period, the following on-site review committee meeting was observed by the inspectors:

Meeting No.	Date
92-130	12/03/92

For the meeting observed, the inspectors concluded that the function of the on-site review committee was effectively implemented.

b. Offsite Review Committee

During this inspection period the inspectors reviewed the licensee's offsite review committee activities which were performed by the nuclear safety review committee (NSRC). To determine if the functions of the committee were being performed in accordance with regulatory requirements, the inspectors reviewed licensee documentation governing the composition, duties

and responsibilities of the NSRC, including section 6.5.2 of the TS. The inspectors reviewed previous NSRC meeting minutes and attended an NSRC meeting to evaluate the effectiveness of the committee to provide an independent review and audit of plant activities.

On December 17, 1992, the inspectors attended the quarterly NSRC meeting. The members were well qualified and prepared to perform the committee reviews. The quorum, composition, and function of the NSRC was in compliance with TS requirements. The NSRC meeting included reviews of various subcommittee reports, including Audit and Quality Assurance, Operations and Maintenance, and Engineering. Discussions were also held concerning planned mid-cycle outage activities and the impact of the fuel leak on plant operation.

The inspectors concluded that the NSRC was objective and effective in the review of plant activities and that the TS requirements for the committee were met.

No violations or deviations were identified.

10. Management Changes

The licensee announced the selection of the Training Manager, Mr. Dave Igyarto as plant manager effective December 7, 1992. Mr. Igyarto succeeded Mr. Robert Stratman, who vacated the position of plant manager to assume the duties of Vice President - Nuclear, upon Mr. Mike Lyster's selection as Vice President at the Dresden Nuclear Power Station.

11. Management Meeting

NRC management met with licensee management on December 17, 1992, at the NRC Region III office in Glen Ellyn, Illinois. Personnel attending the meeting are designated by (#) in paragraph 1 of this report. The purpose of the meeting was to discuss recent plant issues including plant material condition and scope of the January 1993 outage. At the conclusion of the meeting, the NRC management acknowledged the licensee's efforts and planned activities.

12. Items for Which a "Notice of Violation" Will Not Be Issued

During this inspection, certain activities, as described above in paragraph 3.a and 3.b, appeared to be in violation of NRC requirements. However, the licensee identified these violations and they will not be cited because the criteria specified in Section VII.B of the "General Statement of Policy and Procedure for NRC Enforcement Actions," (Enforcement Policy, 10 CFR Part 2, Appendix C, (1992)), were satisfied.

13. Exit Interviews

The inspectors met with the licensee representatives denoted in paragraph 1 throughout the inspection period and on December 28, 1992. The inspectors summarized the scope and results of the inspection and discussed the likely content of the inspection report. The licensee did not indicate that any of the information disclosed during the inspection could be considered proprietary in nature.

During the report period, the inspectors attended the following exit interview:

Inspector	Exit Date
S. Orth	12/10/92