U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-440/92022(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: October 20 through November 20, 1992

Inspectors: A. Vegel
P. Hiland
E. Duncan

Approved By: R. D. Lanksbury, Chief 12/10/92
Reactor Projects Section 3B Date

Inspection Summary

Inspection on October 20 through November 20, 1992 (Report No. 50-440/92022(DRP))
Areas Inspected: Routine unannounced safety inspection by resident inspectors of licensee event report followup, surveillance observations, maintenance observations, operational safety verification, event followup, cold weather preparations, and evaluation of licensee self-assessment capabilities.

Results: Of the seven areas inspected, one violation was identified concerning a failure to enter the Technical Specification (TS) 3.0.3 action statement when the limiting condition for operation could not be met following identification of excessive secondary containment bypass leakage (paragraph 6.b.3). In addition, one non-cited violation (NCV) was identified in the area of licensee event report followup (paragraph 2.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 until October 23 when a reactor shutdown was commenced to repair a crack in the condensate header. On October 24, during the shutdown, the reactor was tripped from 22 percent power due to an inability to complete a surveillance of the rod pattern controller. The plant was restarted on October 30 and remained at or near full power for the duration of the

report period. Operator control of the plant shutdown and startup was good, with the exception of a missed average power range monitor gain adjustment surveillance on November 1.

Maintenance/Surveillance

The quality of observed maintenance and surveillance activities was good. During a tour of the drywell, the inspector identified tools and other debris left behind from previous work activities. An apparent lack of management supervision to ensure work area cleanliness during and after maintenance activities was a significant contributing cause for the deficiencies.

Engineering and Technical Support

The engineering evaluation which supported continued plant operations with the condensate pipe crack was assessed as weak by the NRC staff. Subsequent discussions with the NRC resulted in the licensee shutting the plant down to repair the crack.

Safety Assessment and Quality Verification

The quality of reviewed event reports was acceptable. The on-site review committee was evaluated as effective. On November 5, an apparent weakness in the licensee's understanding of the basis for Technical Specification requirements contributed to the licensee's failure to take action in accordance with TS 3.0.3.

DETAILS

1. Persons Contacted

   a. Cleveland Electric Illuminating Company

      R. Stratman, Vice President - Nuclear, 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
      *G. Cad, Supervisor, Maintenance Section, PNPP
      D. Cobb, Superintendent, Plant Operations, PNPP
      *W. Wright, Manager, Instrumentation and Controls Section, PNPP

   b. U. S. Nuclear Regulatory Commission

      *R. Lanksbury, Chief, Reactor Projects Section 3B, RIII
      P. Hiland, Senior Resident Inspector, RIII
      *A. Vegel, Resident Inspector, RIII
      *E. Duncan, Reactor Engineer, RIII

      * Denotes those attending the exit meeting held on November 20, 1992.

2. 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 50-440/92004-00: On March 21, 1992, during a planned manual shutdown of the plant, a reactor water cleanup (RWCU) system containment isolation occurred as a result of high system differential flow. Immediate corrective action was taken to verify that no actual system leakage had occurred. The RWCU system was secured and subsequently returned to service.

Licensee Investigation of Root Cause and Corrective Actions

Root Cause

The licensee determined the root cause for this event was a design deficiency associated with the unexpected formation of steam voids in the RWCU regenerative heat exchangers (RHXs) while operating in the reduced feedwater temperature mode of operation. As the RWCU system cool down progressed, the voids collapsed and return flow refilled the RHXs instead of returning to the reactor pressure vessel (RPV) via the feedwater injection line. A high differential flow was sensed by the leak detection system and resulted in a RWCU containment isolation.

Corrective Action

Pursuant to corrective actions for previous similar events, the licensee submitted to the NRC on October 30, 1991, a Technical Specification Change Request, "Reactor Water Cleanup System Isolation Actuation Instrumentation." The request included changes to various instrument setpoints to alleviate spurious system isolations. In addition, modifications were made to the computer monitoring program to alert the operators that conditions for steam voiding may exist in the system. All licensed operators were trained on the lessons learned with regard to this event. At the time of this report the proposed Technical Specification change was still under review by the NRC.

Inspectors Review

The initial investigation of this event was documented in Inspection Report 50-440/92003, paragraph 6.b.(5), dated April 27, 1992. A review of licensee efforts in response to similar RWCU isolation events was also documented in the above report in paragraph 3.a. During this inspection period, the inspectors reviewed applicable licensee documentation and noted that all corrective action commitments specific to this event were completed. This item is closed.

b. (Closed) LER 50-440/92011-00: Failure to perform surveillance results in two inoperable intermediate range monitor (IRM) channels and violation of TS 3.3.1.a. On May 3, 1992, while in Operational Condition 5, REFUELING, two IRM channels became inoperable for approximately 10 hours due to failure to complete the surveillance requirements. In addition, the TS 3.3.1.a action requirement to insert a half scram on the reactor protection system (RPS) for the inoperable IRMs was not performed within the 1 hour TS time limit.

Licensee Investigation of Root Cause and Corrective Actions

Root Cause

The licensee determined the root cause of this event was personnel error, failure to follow procedure. When reviewing the surveillance schedule on May 2, 1992, the Unit Supervisor (US) failed to refer to surveillance instruction SVI-C51-T0022C "IRM C and G Neutron Flux Trips Channel Functional Test for 1C51-K601C and 1C51-K601G" to determine TS requirements prior to deferring the surveillance. This review was required by Plant Administrative Procedure (PAP-1105), "Surveillance Test Control."

Corrective Action

Corrective actions included counselling the US with respect to the requirement for procedural compliance. Additionally, 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 the applicable licensee documentation and noted that all corrective action commitments were completed. The inspectors concluded that the licensee's corrective actions appeared reasonable and adequate to prevent recurrence. Failure of the licensee to insert a half scram on RPS Channel C within 1 hour of the IRMs being rendered inoperable was a violation of TS action statement 3.3.1.a. 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.

No violations or deviations were identified; however, one non-cited violation was identified.

3. Monthly Surveillance Observation (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-E32-T0398A	Main Steam Isolation Valve (MSIV) Leakage Control System - Outboard Steam Line Header Pressure Functional For 1E32-N655.
SVI-E22-T1183	High Pressure Core Spray Valve Lineup Verification and System Venting.
SVI-C71-T0039	MSIV Reactor Protection System Channel Check.

No violations or deviations were identified.

4. Monthly Maintenance Observation (62703)

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 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 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 the status of outstanding jobs and to assure that priority was assigned to safety-related equipment maintenance which may affect system performance.

Specific Maintenance Activities Observed:

Work Order/Repetitive Task No.	Title
WO-91-3399	Valve 1N25F0536B - Body to Bonnet Gasket Replacement.
WO-92-4178	Replace Motor on Reactor Water Cleanup Valve 1G33F0004.
WO-92-4023	Replace Motor Pinion Gear Set on Reactor Core Isolation Cooling for Valve 1E51F0064.

WO-92-3115	Install Strainer Basket on Service Water Pump A.
WO-92-3116	Install Strainer Basket on Service Water Pump B.

No violations or deviations were identified.

5. 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 and the emplacement of a new breakwall on site property.

a. Condensate Pipe Crack

On October 19, 1992, the licensee identified a steam leak in the vicinity of the direct contact heater. Subsequent investigation of the leak on October 22, which included removal of insulation, uncovered a crack approximately 9 inches (22.8 cm) in length. The crack was located near a welded connection between the 30-inch (76.2 cm) main condensate header and one of three 18-inch (45.7 cm) lines which fed the direct contact heater. The licensee set up a camera to monitor the crack and initiated plans to temporarily repair the leak with a clamping device. On October 23, following discussions with NRC management concerning the possibility of crack propagation, the licensee commenced a plant shutdown to repair the crack. While shut down, the licensee weld repaired the crack and restored the condensate system back to operation on October 27, 1992. The licensee's root cause determination was in progress at the end of the inspection period.

The inspectors reviewed documentation, observed repair efforts, and discussed with licensee management the significance of the crack. Though the licensee did shut down the plant to repair the leak, the inspectors were concerned with the licensee's initial

plans to only monitor the leak while temporary repairs were being formulated. The inspectors concerns were based on the uncertain cause for the leak and the potential personnel hazard that the leak presented in case a rupture occurred abruptly while repairs were in progress. In addition, NRC regional and headquarters management questioned the licensee's engineering evaluations supporting continued operation until temporary repairs could be affected. Specifically, the licensee's engineering staff appeared to have only accounted for stresses imposed on the crack by normal system pressure, not taking into account other potential stresses including manufacturing and/or vibration induced stresses. Though the licensee did eventually take the conservative action of shutting down, the inspectors were concerned with the licensee's apparently inadequate engineering evaluation which supported continued operation with the crack. The adequacy of licensee engineering evaluations will be further reviewed in future inspections of licensee engineering and technical support activities.

b. Plant Housekeeping and Equipment Condition

During the forced outage to repair the condensate pipe crack, the inspectors conducted walkdowns of areas normally not inspected during power operations. The inspectors toured the drywell, the main steam line tunnel, and other areas in the turbine building and heater bay. As a result, several discrepancies were noted concerning plant housekeeping and equipment condition.

In the drywell, the inspector identified tools, bolts, scaffolding material, and other debris left over from previous maintenance and surveillance activities. The licensee subsequently removed the material from the drywell. The inspector discussed with licensee management what controls were in place to ensure that work areas were maintained clean and that tools were accounted for. Though licensee procedures do provide guidance on post work cleanup and tool accountability, the licensee concluded that an apparent lack of supervisory oversight to ensure that the guidance was being followed was the cause for the discrepancies. Licensee action to improve supervisory performance in this area was being evaluated in a programmatic review of supervisor responsibilities. The adequacy of the licensee's corrective actions to improve post maintenance cleanup will be evaluated in future inspections of plant maintenance activities and during routine observation of plant housekeeping.

While conducting an inspection of the lower elevations of the turbine building, the inspector observed a U-bolt pipe hanger support on the floor. Upon further investigation, the inspector determined that the U-bolt apparently was from a pipe support for the 12-inch (30.5 cm) circulating water box drain tank pump discharge header. Subsequently, upon being notified of the discrepancy by the inspector, the licensee conducted a walkdown of

the system. As a result, the licensee discovered an additional U-bolt missing, weld cracks on several pipe supports, and several loose nuts on Hilti anchor bolts. A nonconformance report (NR-92-N-298) was initiated to evaluate and document the deficiencies and corrective action was initiated to repair the affected supports. The licensee determined the root cause for the failure was a possible system transient due to previous problems with the waterbox drain pump and discharge check valve. The inspector questioned the licensee on why a system walkdown was not performed by the system engineer during the shutdown, since problems with the system were experienced earlier in the operating cycle. As a result of discussions with licensee management, the inspector concluded that due to the workload on the engineering staff during the outage, particularly in support of repairs to the condensate pipe, a system engineer walkdown of the circulating water system was not performed in the turbine building. The licensee determined that improved management of the system engineer workload was required to ensure that system walkdowns were performed. The inspectors will continue to evaluate licensee performance in this area during routine evaluation of the condition of plant equipment.

c. Plant Shutdown and Startup Observations

On October 24, 1992, during the plant shutdown to repair the crack in the condensate header, the operators manually tripped the reactor from 22 percent reactor power. The reactor trip was necessitated by the inability to perform steps 5.1.3.5 or 5.1.4.5 of Technical Specification required surveillance SVI-C11-T1019, "Rod Pattern Controller System Test Below Low Power Setpoint." The steps required that an insequence rod not at an insert or withdrawal limit be selected to verify that block and inhibit indications reset. At the time this surveillance was performed, all insequence rods were at the insert or withdrawal limits. Therefore, the operators could not complete the steps in the surveillance procedure. The operators subsequently tripped the reactor in accordance with plant operating procedures and proceeded to place the plant in a cold shutdown condition. To prevent recurrence, the licensee revised surveillance procedure SVI-C11-T1019 to allow for rods at the withdrawal limit to be tested to verify operability of the rod pattern controller. The procedure was revised on October 28, 1992, prior to the plant restart.

During the plant shutdown, repairs were conducted on the condensate header crack and maintenance was performed to repair various other minor steam leaks in the plant. In addition, the outboard reactor water cleanup and reactor core isolation cooling containment isolation valves were modified to correct deficiencies previously identified in inspection report 50-440/91018(DRS) dated November 5, 1992.

On October 30, 1992, upon completion of maintenance and repair activities, reactor startup was commenced. During the ascension to full power, a problem was identified with feedwater flow indications, and a Technical Specification required surveillance was not performed when required. These problems are discussed separately in paragraphs 5.d and 5.e of this report.

During the plant shutdown and startup, the inspectors observed control room activities. In general, both evolutions were performed in a controlled and deliberate manner. With the exception of the missed surveillance, discussed in paragraph 5.e, the operators performed their tasks in accordance with plant instructions. Overall, the inspectors concluded that the operators remained attentive to changing plant conditions and maintained positive control of the evolutions in progress.

d. Feedwater Flow Anomalies at Low Power

On November 1, 1992, during plant startup, difficulties with feedwater flow instrumentation were experienced which prevented operators from obtaining an accurate heat balance. Specifically, feedwater flow, as determined by the differential pressure across the feedwater venturis was indicating lower than expected at low power based upon independent indications such as steam flow, reactor feed pump turbine (RFPT) suction flows, and megawatts (electric).

The licensee verified that pressure transmitters and associated instrumentation were within calibration limits and reviewed historical data for trending analysis. As a result, the licensee determined that during each startup a higher reactor power level had been required before reactor power based on feed flow (from the venturis) would agree with reactor power based on other indications. In Cycle 1 it was possible to obtain a core heat balance at 15 percent thermal power. During Cycle 2, power had to be increased to 20 percent to obtain a balance; during Cycle 3, to 25 percent; and now in Cycle 4, to greater than 25 percent. During the November 1, 1992, startup, power was at 52 percent when agreement between power indications was obtained. At the end of the inspection period, the licensee was investigating this phenomenon. Though the cause for the indication problems at low power had not been determined, the licensee evaluated that at higher reactor power the flow venturis were apparently providing accurate indication of feed flow. Due to the unknown cause for the indicated flow disparity, the licensee conservatively decided to limit reactor power to 99 percent until a better understanding of the phenomenon could be made. The power limit of 99 percent is based upon an apparent 0.5 psid error in differential pressure across the venturi extrapolated to a .1 Mlbm/hr error at 100 percent reactor power. The inspectors will continue to monitor licensee efforts to determine the root cause for the feed flow venturi indication problem.

e. Missed Average Power Range Monitor Gain Adjustment Surveillance

On November 1, 1992 the licensee identified that the average power range monitor (APRM) gain adjustment surveillance had not been performed as required by TS. Specifically, TS table 4.3.1.1-1 footnote (d) required that APRM channels be calibrated to conform to the power values calculated through a heat balance during Operational Condition 1 when thermal power is greater than or equal to 25 percent of rated thermal power. Additionally, for the provisions of TS 4.0.4 not to be applicable, the surveillance was to be performed within 12 hours of reaching 25 percent power. Technical Specification 4.0.4 prevents entry into an operational condition unless the surveillance requirements associated with the limiting condition for operation (LCO) have been performed within the applicable surveillance interval. During the startup, with the plant in Operational Condition 1 at 24.4 percent power, the licensee completed surveillance SVI-C51-T0024 "APRM Gain and Channel Calibration" on October 31, at 8:12 p.m. (EST), 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. The feedwater flow venturi problem is discussed in paragraph 5.d of this inspection report. 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. Following power increase to 52 percent, SVI-C51-T0024 was completed satisfactorilly at 10:30 p.m.

The licensee initiated condition report CR-92-255 to document event investigation and track subsequent corrective action. In addition, an LER was to be submitted. The inspectors will review the results of the licensee's investigation and evaluate the adequacy of corrective action in a future inspection report.

f. Inoperable Containment Pressure Channel Due to Improper Valve Lineup

On November 8, 1992, while in Operational Condition 1, POWER OPERATIONS, plant operators noticed a discrepancy between the two wide range containment pressure channels displayed on the Emergency Response Information System (ERIS). Containment pressure instrument D23-N270A read -0.3 psig, while the other channel's instrument, D23-N270B, read +0.3 psig. The licensee commenced an investigation. On November 10, 1992, Instrumentation and Control (I&C) personnel discovered that containment pressure instrument D23-N270A was in an improper valve lineup, rendering the instrument inoperable. The licensee initiated condition

report CR-92-261 to document event investigation and track subsequent corrective action. In addition, an LER was to be submitted at a later date.

The inspectors will review the results of the licensee's investigation and evaluate the adequacy of corrective action in a future inspection report.

g. Feedwater Control Problems

On November 11, 1992, at 7:45 p.m., with the plant operating at 99 percent power, a perturbation in the feedwater control system resulted in a reactor water level and power transient. Reactor water level increased to above the level 7 high level alarm setpoint then decreased to below the level 4 low level alarm setpoint before settling out in the normal operating band. Reactor power oscillated for approximately 1 minute then returned to normal. The peak reactor thermal power during the transient was 100.9 percent. Initial troubleshooting determined there was a problem in the "B" reactor feed pump turbine (RFPT) control system, and the controller was placed in manual. On November 12, at 6:35 a.m., with the "B" RFPT in manual, a similar transient occurred. Again reactor water level 7 and level 4 alarms were received and power oscillations peaked at 100.9 percent. Subsequent troubleshooting isolated the cause of the feedwater controller problem to a circuit board in the feed flow logic circuitry. Specifically, the indicated feed flow from the "B" RFPT circuit card apparently spuriously spiked causing RFPT "A" to respond to the false feed flow signal. As a result, the "A" RFPT would initially respond to the false feed flow variance then react to correct the subsequent reactor level changes. The licensee replaced the suspected circuit cards and returned the RFPT to the master level controller on November 13. Following repairs, the feedwater control system operated normally without further problems. The licensee was investigating the root cause of the circuit card problems.

h. Inadequate Retest of Lower Containment Airlock Door

On October 21, the lower containment airlock was declared inoperable for maintenance to investigate seal inflation light indication problems for the inner door. A faulty pressure switch was identified as the cause for the indication problem. The pressure switch was replaced and leak-checked. A retest, which required cycling of the airlock inner door three times while verifying proper operation of the door, lights, and pressure switch was performed. On October 23, at 3:15 a.m. the lower containment airlock was declared operable.

On October 23, at 10:00 a.m., the licensee determined that the retest performed earlier was inadequate because the pressure integrity of the lower containment inner airlock door seal

pneumatic system was broken by the maintenance performed. As a result, TS surveillance requirement 4.6.1.3.e (conducting a seal pneumatic system leak test) was not met. Consequently, the airlock was again declared inoperable. On October 24, at 6:32 a.m., the airlock was declared operable following satisfactory performance of surveillance SVI-P53-T7305, "Lower Containment Airlock Pneumatic System Leak Test, Pen #305."

On November 12, the licensee identified a discrepancy between the piping system diagram, the electrical elementary diagram, and the pressure switch as installed in the field for the lower containment airlock. Further investigation determined that the retest on October 24 was done on the wrong pneumatic system, rendering the retest invalid. As a result, the lower containment inner airlock door was declared inoperable at 1:30 p.m. On November 13, at 6:43 a.m., the test of the seal pneumatic system associated with the pressure switch was completed satisfactorily and the lower containment airlock was declared operable.

The licensee initiated an investigation of this event and documented their findings and corrective actions in LER 92-20 dated November 20, 1992. The inspectors will review the results of the licensee's evaluation and evaluate the adequacy of corrective action during review of LER 92-20 in a future inspection report.

No violations or deviations were identified.

6. 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) Reactor Protection System Actuation

On October 24, 1992, at 12:56 a.m., during the plant shutdown to repair the condensate header crack, following a manual reactor trip from 22 percent reactor power, a reactor protection system (RPS) actuation occurred on low reactor water level. A full scram signal was generated by the RPS when reactor water level decreased below the level 3 setpoint. This event followed a planned manual scram performed in accordance with plant operating procedures. All of the control rods were already inserted and no rod movement occurred due to the second RPS actuation. The licensee informed the NRC Operations Center of this event via the Emergency Notification System (ENS) at about 3:30 a.m. on October 24, 1992.

The licensee conducted a review of this event and determined that the level 3 RPS actuation was an expected plant response. Due to the reactor water level shrink and swell characteristic of a boiling water reactor (BWR), the level decrease was expected and was accounted for by procedural steps to raise water level to the top of the operating band before the scram was initiated. Additionally, operations personnel anticipated the level transient by reviewing appropriate instructions before the manual reactor trip was initiated. The licensee concluded that the water level decrease below the level 3 setpoint was part of a preplanned sequence during reactor operation and need not be reported. On November 13, the licensee informed the NRC Operations Center and retracted the October 24 notification.

The inspectors reviewed licensee procedures, interviewed plant operators, and reviewed data from previous reactor trips. The inspectors concluded that the operators were cognizant of the expected plant transient and that the plant response was similar to previous reactor trips from approximately the same power level.

(2) Local Leak Rate Testing of Nuclear Closed Cooling Containment Isolation Valve

On October 27, 1992, at 11:00 p.m., the licensee determined that containment secondary bypass leakage exceeded Technical Specification 3.6.1.2.d limits following local leak rate testing (LLRT) of nuclear closed cooling (NCC) containment isolation valve 1P43F140. The licensee informed the NRC Operations Center of this event via the ENS at about 2:00 a.m. on October 28, 1992.

During performance of "NCC Containment Isolation Valve Operability Test" surveillance on October 27, outboard containment isolation valve 1P43F140 gave dual indication when in the closed position. Following maintenance, in which the valve position limit switches were adjusted to correct the indication problem, a LLRT was performed. The LLRT was required because 1P43F140 was a butterfly valve whose leakage characteristics could be affected by limit switch adjustments. Excessive leakage was noted and the ENS notification was made. At the time of the notification, it was not known if the maintenance caused the leakage or if the valve had been leaking during operation.

The licensee conducted an investigation and determined that the failure of the post maintenance test did not provide evidence that the valve was leaking during the modes of operation for which the Technical Specification was applicable. Technical Specifiation 3.6.1.2.d applied in Operational Conditions 1, 2, and 3. At the time of the post maintonance test, the plant was in Operational Condition 4, where the leakage requirements were not applicable. The licensee determined that the cause for the excessive leakage was the maintenance activity performed on the limit switches and that there was no firm evidence to indicate that the valve was leaking during the modes of operation for which the Technical Specifications were applicable. Based on the above, on November 13 the licensee informed the NRC Operations Center and retracted the October 27 notification.

(3) Secondary Containment Bypass Leakage in Excess of Limits

On November 5, 1992, while in Operational Condition 1, POWER OPERATIONS, at 99 percent reactor power, the licensee discovered that based upon the results of local leak rate testing (LLRT) of a 42-inch (106.68 cm) containment purge supply line penetration, the secondary containment bypass leakage rate exceeded TS limits. Upon recognizing that containment bypass leakage limits were exceeded, the licensee failed to take appropriate action as specified in TS 3.0.3.

On November 5, at 4:15 p.m., upon reviewing the data obtained during performance of SVI-M14-T9313, "Type C Local Leak Rate Test of 1M14 Penetration V313," a leakage rate of 3130 standard cubic centimeters per minute (sccm) was calculated for the penetration. Specific to this penetration, TS 4.6.1.8.4 allowed leakage up to .05 L_a (5011.6 sccm). Therefore the penetration leakage was within limits. However, when the leakage from the penetration (3130 sccm) was combined with the previously recorded secondary containment bypass leakage, the total,

6598.34 sccm, was greater than the total secondary containment bypass leakage limit of .0504 L_a (5051.74 sccm) allowed by TS 3.6.1.2.d. The subsequent action statements applicable in Operational Conditions 1, 2, and 3 required, in part, that secondary containment bypass leakage paths be restored to less than or equal to .0504 L_a prior to increasing reactor coolant temperature above 200 °F. Since the plant was in Operational Condition 1 at 99 percent reactor power, the licensee's staff questioned the applicability of the TS in their current plant condition. Though the licensee did recognize that they might be in a condition outside of the TS LCO, and therefore TS 3.0.3 would be applicable, no action was taken in accordance with the TS 3.0.3 requirement. At approximately 7:00 p.m. the licensee contacted a member of the NRC staff to discuss the situation. At that time it was not apparent that the TS 3.0.3 requirements were applicable. On November 6, 1992, at approximately 6:00 a.m., the containment purge supply penetration was declared operable following valve repair and satisfactory retest. After a review of TS requirements by the NRC staff and discussions between the staff and the licensee on November 6 it became apparent that the licensee did not take appropriate action in accordance with TS 3.0.3. On November 6, at 6:30 p.m., the licensee notified the NRC Operations Center via ENS that required actions in accordance with TS 3.0.3 were not taken on November 5 following discovery of excess secondary containment bypass leakage.

Technical Specification 3.0.3 required that when a limiting condition for operation (LCO) is not met, except as provided in the associated action requirements, within 1 hour action shall be initiated to place the unit in an operational condition in which the specification does not apply.

Contrary to the above, on November 5, 1992, upon not meeting the LCO action requirements of TS 3.6.1.2, the licensee failed to initiate action within 1 hour, in accordance with TS 3.0.3, to place the plant in Operational Condition 4, COLD SHUTDOWN, where TS 3.6.1.2 would not be applicable. This is a violation (440/92022-01(DRP)).

One violation and no deviations were identified.

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

a. On-Site Review Committee

During the report period, the inspectors observed on-site review committee meetings to evaluate that organization's 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 (USAR), or vendor technical manuals for their determinations in addition to the Technical Specifications.

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
92-128	11/19/92

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

No violations or deviations were identified.

8. Cold Weather Preparations (71714)

The inspectors reviewed the licensee's implementation of the freeze protection program. This included a review of freeze protection off-normal instruction ONI-R36 and various system operating instructions, including those for emergency service water, condensate storage and transfer, and heat tracing. A walkdown of heat trace control panels and other freeze protection systems to verify proper operation was performed. Also, the inspectors discussed implementation of the freeze protection program with operations and engineering personnel to assess their familiarity with cold weather preparations.

Off-normal instruction ONI-R36 and pertinent system operating instructions appeared to provide adequate freeze protection when implemented. The portions of the freeze protection systems walked down by the inspectors were observed to be operating properly, and plant personnel questioned were familiar with ONI-R36 and its contents. The inspectors noted that although heat tracing had not been reinstalled following the circulating water system pipe rupture event on December 22, 1991, contingency plans to prevent this piping from freezing in the event of a system shutdown appeared adequate. The inspectors also noted that specific procedures to address freeze protection in the event of a failure of the freeze protection system did not exist. However, discussions with operations and engineering personnel indicated that

contingency plans had been considered and appeared reasonable. Based upon the above observations, the inspectors concluded that the licensee's freeze protection program had been adequately implemented.

No violations or deviations were identified.

9. Management Changes

The licensee announced that Mr. Mike Lyster, Vice President - Nuclear, the senior licensee manager on site, resigned his position effective December 17, 1992. Mr. Lyster will become the site vice president at the Dresden Nuclear Power Station. At the end of this inspection period the licensee had not announced his successor.

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

During this inspection, certain activities, as described above in paragraph 2.b, appeared to be in violation of NRC requirements. However, the licensee identified this violation and it 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.

11. Exit Interviews

The inspectors met with the licensee representatives denoted in paragraph 1 throughout the inspection period and on November 20, 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 interviews:

Inspector	Exit Date
J. R. Kniceley (Security)	11/20/92