U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-440/91004(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: February 28 through April 17, 1991

Inspectors: G. O'Dwyer

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Approved By: R. D. Lanksbury, Chief
Reactor Projects Section 3B

5/6/91
Date

Inspection Summary

Inspection on February 28 through April 17, 1991 (Report No. 50-440/91004(DRP))

Areas Inspected: Routine unannounced safety inspection by resident inspectors of previously identified items, reactor and computer engineering, licensee event report followup, monthly surveillance observations, monthly maintenance observations, operational safety verification, onsite followup of events, and allegation review.

Results: Of the eight areas inspected, no violations were identified in five areas; one violation was identified in each of the areas of licensee event report followup (paragraph 4), event followup (paragraph 8.b.(5)), and allegation followup (paragraph 9). However, in accordance with 10 CFR Part 2, Appendix C, Section V.G.1, a Notice of Violation was not issued.

For this report period, the functional area of plant operations was considered good. Examples of good operator response included response to a rod drift event, prompt plant shutdown due to increased unidentified leakage, and controlled plant restart at the conclusion of the inspection period.

The area of maintenance and surveillance was considered good during the report period. In addition to the observed maintenance and surveillance test activities, the licensee's response to a check valve surveillance test failure

was prompt and in accordance with Technical Specification. One negative item in the maintenance area was the repairs performed and documented on control room ventilation ductwork (NCV 440/91004-01 and -03(DRP)).

Engineering and Technical support was considered good. The evaluations for diesel test failures were prompt and detailed. The support to resolve problems in weld profiles was good. Of particular note was the effort to determine root cause for a Reactor Core Isolation Cooling system check valve failure.

The area of radiological controls was considered adequate and improving. Plant housekeeping was considered adequate.

The area of Safety Assessment and Quality Verification was considered good. The licensee self-identified several violations during the report period for which prompt and appropriate corrective actions were taken.

In general, the inspectors found the areas of security and emergency preparedness to be a strength based on routine observations.

DETAILS

1. Persons Contacted

   a. Cleveland Electric Illuminating Company (CEI)

      R. Stratman, General Manager, Perry Nuclear Power Plant (PNPP)
      *M. Gmyrek, Operations Manager (PNPP)
      M. Cohen, Manager Maintenance Department (PNPP)
      D. Cobb, Operations Superintendent (PNPP)
      S. Kensicki, Director, Perry Nuclear Engineering Department (PNED)
      V. Concel, Manager, Technical Section (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 (PNAD)
      W. Wright, Acting Manager, Instrumentation and Controls Section (PNPP)
      *E. Root, Manager, Performance Engineering (PNED)
      *J. Eppich, Manager, Mechanical Design (PNED)

   b. U. S. Nuclear Regulatory Commission

      *P. Hiland, Senior Resident Inspector, RIII
      *G. O'Dwyer, Resident Inspector, RIII
      S. Stasek, Senior Resident Inspector, RIII
      F. Brush, Resident Inspector, RIII

   * Denotes those attending the exit meeting held on April 17, 1991.

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

   a. (Closed) Open Item (440/90002-01(DRP)): Immediate actions after a reactor scram which bypassed the low pressure Main Steam Line Isolation Valve (MSIV) closure signal. This item concerned an issue identified at another boiling water reactor (BWR) when manual actions taken following a reactor scram bypassed an automatic MSIV closure signal.

      As documented in licensee Condition Report 90-023, approved April 27, 1990, automatic or manual closure of the MSIVs was insignificant to the overall consequences of a pressure regulator fail-open transient described in the Updated Safety Analysis Report (USAR) Table 15.1-4. However, the investigation performed identified multiple scenarios associated with the pressure regulator failure that were not clearly or completely described in the USAR. As a result, two corrective actions were initiated. The first action was to review USAR Chapter 15 transient sequence of events against plant off-normal instructions. That action was completed on May 4, 1990, and resulted in nineteen comments requiring resolution. The second action item was assigned to the licensee's engineering and licensing departments to resolve inconsistencies in transient sequence analysis. That action item was tracked as Item No.

90-023-2 in the licensee's commitment tracking system. The inspectors discussed the current status of the licensee's review with cognizant personnel.

Based on the completed analysis documented in Condition Report 90-023, completion of Action Item 90-023-1, and the current schedule of Action Item 90-023-2, the inspectors have no further questions on this subject. This item is closed.

b. (Closed) Open Item (440/89022-01(DRP)): Utilization of the quarterly work schedule. As detailed in the Diagnostic Evaluation Team (DET) report for the Perry Nuclear Power Plant dated May 1989, Sections 3.2.5 and 3.6.2, a number of equipment problems were identified during the first operating cycle. In response, the licensee planned to utilize their quarterly work schedule to maintain the backlog of corrective maintenance low.

Since the DET report was issued in May of 1989, the inspectors have monitored the licensee's maintenance backlog. In addition, a maintenance team inspection was performed in September and October 1990, the results of which were documented in Inspection Report 50-440/90-012(DRS) dated November 14, 1990. Since the DET report was issued, the licensee has maintained a reasonable backlog of corrective maintenance work orders. At the time of this report, the non-outage backlog was 502 with less than 50 percent greater than three months old. That was an increase over the 172 backlog noted during the maintenance team inspection. However, the increased backlog appeared reasonable since the emphasis was placed on completing outage work orders in the second refuel outage ending January 1991. Since that time, the backlog was reduced about 20 percent.

Based on the licensee maintaining the corrective maintenance backlog at a reasonable level over the past two years, the inspectors concluded that the quarterly work schedule was effectively utilized. This item is closed.

c. (Closed) Violation (440/89022-07(DRP)): Failure to provide adequate instructions to assure positioning of throttled control valves. As detailed in the Diagnostic Evaluation Team (DET) report for the Perry Nuclear Power Plant dated May 1989, Sections 2.1.2.6 and 3.2.3, the licensee failed to provide proper administrative controls for throttled valve positioning.

The licensee responded to the DET finding of inadequate throttle valve control in letter PY-CEI/NRR-1043L, dated July 29, 1989. As detailed in that response, the licensee implemented a locked throttle valve program which included independent verification. The system valve lineups (VLIs) were revised and the surveillance instructions (SVIs) were to be revised to assure system restoration checklists restored throttled valve positions.

During this report period, the inspectors reviewed licensee Surveillance Report No. 89-340, dated December 8, 1989. In

addition, the inspectors reviewed several valve lineup instructions and surveillance instructions and noted the throttled valves to be identified. Based on completion of the corrective action as stated in the licensee's response to the subject violation, and the inspectors verification by direct field observation that controls on throttled valve positioning were maintained, this item is closed.

d. (Closed) Open Item (440/89022-10(DRP)): Equipment trend analysis. This item was previously reviewed by the inspectors as documented in Inspection Report 50-440/90005(DRP), Paragraph 2.d, dated May 4, 1990. At the conclusion of that inspection period, this item remained "open" pending the inspector's review of the action taken to resolve audit Action Request (AR) PA90006-001. That action request concerned a lack of administrative controls over the reliability information tracking system (RITS).

During this report period, the inspectors reviewed the approved resolution to AR PA90006-001, dated December 12, 1990. As stated in that resolution, Revision 2 to Plant Administrative Procedure (PAP)-1601, "Failure Analysis" was effective December 10, 1990. Section 6.5 of PAP-1601 detailed the administrative controls over the trending analysis (i.e. RITS) effort. Based on resolving the outstanding action request to establish administrative controls and the actions taken by the licensee as previously documented in Inspection Report 50-440/90005, this item is closed.

e. (Closed) Open Item (440/89022-12(DRP)): Check valve testing. As detailed in Section 3.4.1 of the Diagnostic Evaluation Team (DET) report dated May 1989, the inspectors noted that Check Valve E51-F011 was not required to be tested in the "alternate position verification" (AP). As noted by the inspectors, the licensee was revising their inservice test program at the time of the DET.

During this report period, the inspectors noted that the licensee included an "AP" test for Check Valve E51-F011 in their inservice test program (ISTP) submitted in accordance with Generic Letter 89-04. As detailed in the licensee's submittal letter PY-CEI/NRR-1063 L, dated October 3, 1989, Perry ISTP Revision 2, Page 3-140 identified the subject "AP" test requirement during refueling outages. The inspectors noted that final acceptance of the Perry ISTP was still under NRC staff review; however, the licensee was responding directly to the NRR Project Manager for final acceptance of the Perry ISTP. This item is closed.

No violations or deviations were identified.

3. Inspection of Reactor Engineering and Computer Engineering (61702, 61705, 61706)

a. General

During this report period, the inspectors reviewed the activities of the licensee's Performance Engineering Section. Included in that

organization were the Reactor Fuel Management Unit and the Computer Engineering Unit. As discussed below, the inspectors reviewed current administrative controls, recent surveillance test results, and control of computer software. In addition, the inspectors discussed the activities being reviewed with the cognizant Unit Lead Engineer.

b. Details

The inspectors reviewed the following instructions and administrative controls: Surveillance Instruction (SVI)-C51-T5351, "LPRM Calibration," and the results of a February 28, 1991, local power range monitor (LPRM) calibration; Plant Administrative Procedure (PAP)-0506, "Computer Access and Software Control;" and PAP-1701, "Plant Records Management." In addition, the inspectors reviewed: the licensee's Quality Assurance Plan, Appendix D, "Computer Software;" computer program and/or modification request for the data update for nuclear steam supply (NSS) programs for Cycle 3 operation; Updated Safety Analysis Report (USAR) Section 4.1; and USAR Appendix 15B.

The LPRM calibration results and the process computer modification data appeared to be complete and correct. The computer access and software control procedure required that a software tracking folder be established for each computer and/or software system (subsystem as appropriate). That procedure further stated that the software tracking folder shall contain a list of the Category I, II, III software programs and a software system identification matrix entry form. The tracking folder for the Monicore and Midas software did not contain these items.

Validation and verification (V&V) of software was required by PAP-0506. There was no "formal" V&V program to ensure that appropriate actions were documented when modifying Monicore software and databanks. The USAR section 4.1, paragraph 4.1.2.1.3, stated that a discussion of the fuel designs utilized for the current cycle were contained in Appendix 15B, "Reload Safety Analysis." The inspectors noted that Appendix 15B contained the reload analysis for Cycle 2. The licensee stated that the current Cycle 3 reload analysis would be inserted as USAR Appendix 15B in March 1992.

No violations or deviations were identified

4. Licensee Event Report Followup (90712)

Through review of records, the following event report was reviewed to determine if reportability requirements were fulfilled, immediate corrective actions were accomplished in accordance with Technical Specifications, and corrective action to prevent recurrence had been established.

(Closed) LER 50-440/90020-00: Maintenance activities on the control room emergency recirculation system resulted in a Technical Specification Violation and compromise of a safety system.

On August 31, 1990, while the reactor was operating at about 96 percent power, a post-maintenance visual inspection of repairs on a control room ventilation and air conditioning (CRHVAC) system backdraft damper revealed a rectangular hole had been cut in the duct downstream of the damper. Temporary repairs were made to the duct and the system was considered operable. On September 1 licensee personnel determined that both trains of the emergency recirculation mode of the CRHVAC system were inoperable because the rectangular hole could have allowed excessive contaminants into the control room after a design basis accident. Technical Specification 3.0.3 was entered. Licensee personnel also concluded that any maintenance or inspections which required the removal of access panels in that portion of the CRHVAC duct (which was a part of the common plenum) caused the inoperability of both CRHVAC trains. Maintenance and inspections had been performed on a routine basis since the beginning of plant operation.

Licensee Evaluation of Cause and Corrective Actions

Root Cause:

The cause of the rectangular hole was an incorrect maintenance activity. The licensee's failure to declare both trains of the emergency recirculation mode of the CRHVAC system inoperable when access panels were removed was due to plant personnel not recognizing that the access panels were in common ducting.

Corrective Actions:

- The hole was repaired and personnel responsible for cutting the hole were counseled.

- Maintenance activities and inspections which compromised the control room envelope were being rescheduled to be performed during plant outages. Applicable procedures were being revised to use additional dampers for isolating control room emergency recirculation system plenums prior to opening plenum access panels.

Inspectors' Evaluation:

The inspectors concluded that the licensee had performed a prompt evaluation of the causes of this event with appropriate management attention. The corrective actions taken appeared reasonable to prevent recurrence.

Failure of the licensee to maintain operable at least one train of the emergency recirculation mode of the CRHVAC system was a Violation of Technical Specification 3.7.2 (NCV 440/91004-01(DRP)). This violation was a "licensee identified item" which met the test of 10 CFR Part 2, Appendix C, Section V.G.; therefore, a Notice of Violation will not be issued. This item is closed.

No deviations were identified. One violation was identified for which a Notice of Violation was not issued.

5. Monthly Surveillance Observation (61726)

For the below listed surveillance activities 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 and 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 Test No.	Activity
SVI-M16-T0417B	Post-LOCA Drywell/Containment Differential Pressure Vacuum Breaker.
SVI-D17-T8002	Liquid Radwaste to Emergency Service Water Radiation Monitor Channel Calibration, D17-K606.
SVI-B21-T0368	Safety Relief Valve Tail-Pipe Pressure Switch Channel Functional/Calibration, 1B21N419A-V.

No Violations or Deviations were identified.

6. 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 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 orders (WOs) 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.

The following specific maintenance activities were observed:

WO	Subject
90-05100	Replaced overload relay for the "B" chiller lubricating oil pump (OP47C5011B).
91-00771	Replaced the hydraulic fluid pump motor on control valve OP47F85B for the control room ventilation system.
91-1663	Division 1 emergency diesel relay removal.
91-1658	Division 2 emergency diesel relay (K-1) replacement.
91-255	Retorqued mounting bolts for the Division 1 emergency diesel generator vibration detectors (1R43N0701A, 2A, 3A, and 4A).
91-1773	Replaced Rosemount transmitter for the Emergency Service Water (ESW) outlet flow from the Division 1 emergency diesel.
91-2119 and 91-2121	Field weld recirculation loop vent valves on 23A and 60A.
91-2185	Reactor core isolation cooling (RCIC) check valve F066 repair.

Regarding Work Order 91-2185, the inspectors discussed with cognizant licensee personnel actions taken to identify the root cause of the reactor core isolation cooling (RCIC) "testable" check valve failure. As documented in licensee Condition Report 91-082, dated April 3, 1991, RCIC "testable" check valve F066 was found stuck open during required COLD SHUTDOWN surveillance testing.

In order to make repairs, the reactor vessel head "flanged piping" was removed and the F066 check valve was disassembled to determine the cause of failure. In addition, the F066 check valve manufacturer, Rockwell, provided a factory representative during the valve inspection process. The inspectors discussed with cognizant licensee personnel the results of their investigation documented in a April 17, 1991, "draft" report that was to be incorporated into Condition Report 91-082. Since the check valve closed during initial disassembly, the exact "as-found" condition was unknown. Attempts to move the check valve disc by hand were successful. The disc stuck open on several occasions and a slight amount of force was required to free the disc.

As detailed in the "draft" report, the licensee documented all anomalous attributes noted during the check valve inspection. Each anomaly was evaluated to determine what affect, if any, it could have on check valve operation. At the conclusion of this report period, the licensee had not

positively identified the root cause or causes for the RCIC F066 check valve sticking in its open position. However, the licensee concluded "the most probable root cause was a combination of new parts (installed during the second refuel outage) in the valve introducing new dimensions and tightening component tolerances." As corrective action the following replacement components were installed: Kalrez seal; silver seal; piston locking pin; and retaining ring locking pin. In addition, all components were cleaned and refurbished. The valve was reassembled under the guidance of the valve manufacturer's representative which included monitoring the installation of the piston head, pin, pressure seal cover, pressure seal gasket, and torquing. RCIC check valve F066 was successfully stroke tested prior to pressurizing the reactor at the end of the forced outage.

No violations or deviations were identified.

7. Operational Safety Verification (71707)

a. General

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 for example, diesel generators - Divisions 1, 2, and 3; low pressure coolant injection system - Division 3; reviewed tagout records, and verified tracking of Limiting Conditions for Operation associated with affected components. Tours of the electrical switchgear rooms - Division 1 and 2; intermediate, auxiliary, containment, drywell, 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.

Through most of this report period the Perry plant was operated at 100 percent power. However, on April 1, increased drywell unidentified leakage required a forced outage to repair failed components. The specific details are discussed below in Paragraph 8.b.(3). At the conclusion of the report period on April 17, the licensee had completed repairs to the failed components and was in the process of returning to 100 percent reactor power.

b. Control Rod Drift

On March 15, 1991, at about 12:00 noon, Control Rod 42-39 was observed "drifting" in from its withdrawn position. In response to the rod drift, plant operators fully inserted Control Rod 42-39 and

directed troubleshooting be commenced to identify the cause for the rod drift. In addition to the single rod drift, eleven other control rods were made inoperable with the rod control and information system (RCIS) in the manual test mode.

Plant operators verified a control rod drive pump was running by inserting Control Rod 42-39 as required by Technical Specification 3.1.3.3. With more than eight control rods declared inoperable, Technical Specification 3.1.3.1 was entered and the licensee made preparations to place the reactor plant in HOT SHUTDOWN within 12 hours. However, prior to reducing reactor power, the cause for this event was identified to be a failed "transponder" circuit board in the electrical control circuit for Control Rod 42-39. That "transponder" circuit board was replaced and the RCIS was returned to a normal configuration. At about 1:45 p.m. all control rods were declared operable. The inspectors concluded the licensee's response to this event was in accordance with Technical Specification requirements.

c. Main Steam Isolation Valve Slow Closure Test

On March 24, 1991, at about 2:30 a.m., while operating at 100 percent reactor power, two main steam isolation valves (MSIVs) failed to respond to a slow-close signal within a specified time. At the time of this event, plant operators were performing a channel functional surveillance test of the MSIV valve closure logic. The purpose of the planned test was to verify receipt of reactor protection system (RPS) signals when the MSIVs were closed about 8 percent from full open. As detailed in Surveillance Instruction (SVI)-C71-T0039, "MSIV Closure Channel Functional," the movement of MSIV F022C and F028C was to be limited to 22 and 25 seconds, respectively, to prevent an inadvertent closure of the MSIVs. During the test the slow-closure stroke of the two MSIVs did not result in receipt of the RPS test signal within the prescribed time limit. The test was suspended and considered a failed surveillance.

The licensee initiated a troubleshooting work order that performed a full closure in both the slow-close and fast-close mode of operation. The fast-close test verified closure times were within Technical Specification limits of 3-5 seconds. The slow-close test was successful in full stroking the MSIVs; however, the time required to reach the 8 percent RPS limit switch was about 3 seconds longer than anticipated in the governing surveillance instruction.

The use of a specific slow-close time limit was corrective action by the licensee to an earlier event documented in Inspection Report 50-440/90014, Paragraph 8.b.(2), dated August 16, 1990. During that event, a MSIV under test was slow-stroked well beyond the RPS limit switch setpoint of 8 percent, in part, due to a lack of understanding of the time required to slow stroke the MSIVs. After reviewing historical data, the 22 and 25 second time limits were incorporated into the surveillance instruction.

After review of the troubleshooting work order, the licensee increased the allowable slow-stroke times for MSIVs F022C and F028C to 28 and 31 seconds, respectively. Subsequently, the channel functional test was successfully performed. The inspectors noted that the initial test performance was conducted in accordance with the test procedure and that test failure was promptly noted by the test performers. The use of a troubleshooting work order was appropriate and verification of the Technical Specification fast-close time limits was promptly performed. In addition, the inspectors considered the licensee's increase in the slow-close time limits to be reasonable based on the number of variables that could affect slow-closure time (i.e. temperature, actuating air pressure, condition of MSIV stem lubricant, etc.).

d. Control of Equipment Tagging

On March 28, 1991, during a routine walkdown of the Division 1 switchgear room, the inspectors noted an out-of-service tag (No. 6047-4) was placed on motor control center (MCC) EAB07-XP (RHR A heat exchanger ESW inlet valve). The tag specified the MCC switch position to be "off/open." However, the MCC was actually "on/energized." Subsequent review and discussions with licensee personnel determined the subject valve was normally open and that if closed, may not be capable of reopening. The licensee had previous knowledge of this longstanding problem and had, as part of their followup action, placed the out-of-service tag on the MCC. The inspectors noted that the out-of-service tag system at Perry was similar to "Caution" tag systems at other facilities. The licensee further indicated plans were to maintain the MCC energized and that the tagging inconsistency would be addressed. The following week, the inspectors noted that the out-of-service tag was removed.

e. Underwater Vacuum Cleaner Storage

During a tour in Containment on April 11, 1991, the inspectors noted that the underwater vacuum cleaner was stored in close proximity to the suppression pool make-up system gravity drain suction nozzles. The concern with that storage location was the potential blockage the stored vacuum cleaner hose could have on the make-up flow requirements. When notified of the inspectors observation, the licensee initiated Condition Report 91-090, dated April 14, 1991. That report identified that the vacuum cleaner had not been stored in its designed storage location (Figure 9.1-27 of USAR) since the first refueling outage in 1989.

The underwater vacuum cleaner was placed in its proper storage location and Field Clarification Request (FCR) 15144, dated April 15, 1991, was written to evaluate the safety significance of the improper storage. That evaluation concluded that, based on the low flow velocities during the make-up system operation and the weight of vacuum cleaner components, no entrainment of components, and therefore no blockage of the make-up system flow path would have occurred. The inspectors noted that the licensee took prompt action to correct the storage deficiency and that the evaluation of no safety impact appeared reasonable.

No violations or 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 licensee investigation had identified root causes of equipment malfunctions and/or personnel errors and were taking or had taken appropriate corrective actions. Details of the events and licensee corrective actions noted during the inspector's followup are provided in paragraph b. below.

b. Details

(1) Loss of Control Complex Chillers

On March 5, 1991, at about 6:00 p.m., while operating at 100 percent power, the licensee entered the provisions of Technical Specification 3.0.3 following a loss of both the "A" and "B" control complex chillers. At the time of event occurrence, the "B" chiller had been removed from service for planned maintenance activities when the "A" chiller tripped on low refrigerant temperature.

Technical Specification 3.7.2, "Control Room Emergency Recirculation System," did not allow continued plant operation with less than one operable control room emergency recirculation subsystem. In accordance with Technical Specification 3.0.3, the licensee made plans to commence a plant shutdown. However, at about 11:00 p.m. the "A" control complex chiller was returned to an operable status before an actual power reduction occurred. The cause for the loss of the "A" chiller was a failed purge valve sticking open allowing the refrigerant to escape. The short term corrective action was to isolate that automatic purge valve, recharge the chiller with refrigerant, and provide instructions to manually purge the chiller.

The licensee notified the NRC Operations Center of this event via the Emergency Notification System (ENS) at about 10:00 p.m. on March 5, 1991.

(2) Loss of Emergency Diesel Generator

On March 14, 1991, at about 2:30 p.m., while operating at 100 percent reactor power, the Division 1 and 2 emergency diesel generators (EDGs) were declared inoperable following surveillance test failures. In accordance with planned monthly test activities the Division 2 EDG was started at about 9:15 a.m.; however, the voltage output on the generator remained at zero. In accordance with Technical Specifications, the Division 1 and 3 EDGs were to be tested within 24 hours to verify operability. The Division 1 EDG was started at about 2:30 p.m.; however, that EDG failed to respond to governor speed control signals. The Division 3 EDG was successfully tested at about 8:30 p.m.. With both the Division 1 and 2 EDGs inoperable, Technical Specification 3.8.1.1 required the plant to be placed in HOT SHUTDOWN within the next 14 hours.

The cause for the Division 2 EDG failure was a degraded field-flash relay (K-1). That relay was replaced with an identical field-flash relay removed from Division 1 EDG switchgear. Following successful post-maintenance testing, the Division 2 EDG was declared operable and the HOT SHUTDOWN Action statement was exited at about 2:30 a.m. on March 15. No actual power reduction was performed. The licensee concluded that the Division 2 EDG failure was a "valid test failure" and submitted the required special report as LER 91-009 dated April 15, 1991.

The cause for the Division 1 EDG failure to respond to speed governor control could not be determined. However, during troubleshooting efforts, cleaning and exercising of the "suspect" speed control components occurred. As stated in LER 91-009, the licensee believed that the conditions preventing proper speed control were corrected during the maintenance activity. The licensee determined that, in accordance with Regulatory Guide 1.108, the Division 1 test failure was not valid since the EDG would not require bus synchronization to perform its intended function. The inspectors reviewed the actions taken by the licensee and concluded that the determinations made were reasonable. The Division 1 EDG was restored to an operable condition at about 11:00 p.m on March 15.

The licensee informed the NRC Operations center of this event at about 4:00 p.m. on March 14, 1991.

(3) Plant Shutdown Due to Increased Unidentified Drywell Leakage

On April 1, 1991, the licensee initiated a plant shutdown to investigate the cause for increased unidentified leakage in the drywell. In order to investigate the source of the increased drywell leakage, the licensee planned to reduce power below 20 percent and visually inspect for leakage inside the drywell.

(A) Sequence of Events

April 1

6:40 p.m.	Based on unidentified drywell leakage increasing to 1.8 gpm, the licensee planned to reduce power to less than 20 percent.
7:00 p.m.	Unidentified drywell leakage at 2.0 gpm.
8:00 p.m.	Started power reduction.
9:00 p.m.	Reactor power at 65 percent, unidentified leakage at 2.5 gpm.
10:20 p.m.	During the power decrease, reactor recirculating pump "A" was unable to downshift to slow speed due to a failure in its associated low frequency motor-generator power source. That failure resulted in the loss of the "A" reactor recirculating pump and the plant entered single-loop operation at 22 percent reactor power. Unidentified drywell leakage was at 4.8 gpm. The licensee entered the 12 hour shutdown action statement of Technical Specification 3.4.1.1 due to single-loop operation.
10:40 p.m.	Decision made to place the plant in HOT SHUTDOWN.
11:00 p.m.	Unidentified drywell leakage peaked at 5.0 gpm.
11:30 p.m.	Notification made to NRC Operations Center of plant shutdown initiation.

April 2

2:05 a.m.	Main generator offline.
2:40 a.m.	Plant entered Mode 2, STARTUP.
4:10 a.m.	Reactor power at 0 percent.
5:00 a.m.	Visual inspection of the drywell identified the source of leakage to be a 3/4 inch vent line on reactor recirculation loop suction isolation valve 1B33-F023B. Decision made to go to COLD SHUTDOWN.

6:58 a.m. All control rods inserted. Plant entered Mode 3, HOT SHUTDOWN.

5:10 p.m. Plant entered Mode 4, COLD SHUTDOWN.

(B) Inspector Review

The inspectors noted that the licensee's decision to reduce reactor power and investigate the increased unidentified drywell leakage was made within a reasonable time of the initial increased leakage. The decision to reduce power to less than 20 percent was intended to permit a visual inspection inside the drywell; however, following the failure of the "A" reactor recirculation pump to downshift to slow speed (see paragraph 8.b.(4). below for details), the licensee was required by Technical Specification 3.4.1.1 to achieve Mode 3, HOT SHUTDOWN within 14 hours of the recirculating pump trip. The inspectors noted that the licensee complied with that requirement.

With the plant at zero percent power in Mode 2, STARTUP, the licensee identified the source of unidentified leakage to be a failed 3/4 inch vent line (at 5:00 a.m. on April 2). Although the licensee placed the plant in Mode 4, COLD SHUTDOWN within the Action requirements of Technical Specification 3.4.3.2 (i.e. achieved COLD SHUTDOWN in 12 hours versus the Action statement time limit of 36 hours), the licensee did not consider the vent valve leak location to be "pressure boundary leakage." The licensee stated that the vent valve location was isolable; therefore, the leak did not meet the Technical Specification definition of "pressure boundary leakage."

Based on the plant conditions at the time of leak discovery (i.e. Mode 2 STARTUP, zero percent power, leakage rates below Technical Specification limits) and the decision to promptly proceed to Mode 4, COLD SHUTDOWN, the inspectors concluded that the licensee operated the plant in a safe manner. However, the inspectors noted that the declaration of an Unusual Event would have been required by the licensee's emergency plan had the specific reactor coolant leakage path been unisolable or if it could not have been determined if it was isolable.

The root cause for the 3/4 inch vent valve failure was still under investigation at the end of this report period. The licensee was aware of industry information that suggested a possible cause for failure. General Electric Service Information Letter (SIL) No. 512, dated May 2, 1990, "Recirculation Loop Unisolable Leakage," discussed the potential for failure of small vent and drain lines connected to the recirculation system. As detailed in SIL No. 512, differential movement between the

system piping and small unsupported vent, drain, or instrument lines concentrates stresses at the heat affected zone which can, over time, lead to fatigue damage. In response to the information provided by SIL No. 512, the licensee performed a liquid penetrant examination of three of the thirty-two vent, drain, and instrument lines during the second refueling outage. The inspectors noted that the three welds inspected during the refuel outage did not include the subject failed weld.

As immediate corrective action, the licensee examined by liquid penetrant exam, the other thirteen unsupported vent and drain lines and found no indications. In addition, the failed vent line was plug welded in accordance with the recommendations of SIL No. 512. The licensee stated that the remaining eighteen small recirculation loop line attachments were all supported instrument lines not susceptible to fatigue damage. As discussed below in sub-paragraph (5), the licensee identified a weld profile deficiency on the fourteen vent and drain connections. Those weld profile deficiencies were not considered to be the cause for the failure experienced.

(4) Loss of Reactor Recirculation Pump A

On April 1, 1991, at about 10:30 p.m., while lowering reactor power to less than 20 percent, the "A" reactor recirculation pump tripped off during a planned downshift from FAST to SLOW speed. At the time of this event plant operators were reducing reactor power to allow drywell entry to investigate the increased unidentified leakage discussed above in sub-paragraph (3). After loss of the "A" reactor recirculation pump, the planned downpower was continued to Mode 3, HOT SHUTDOWN.

Troubleshooting identified that the cause for the failure of the "A" reactor recirculation pump to downshift was a degraded field-flash relay in the associated low frequency rotor-generator (LFMG). The degraded relay was replaced and the reactor recirculation pump was successfully started in SLOW speed at about 7:45 p.m. on April 2, 1991. The licensee's investigation into the "root cause" for this event was to be documented in Condition Report 91-078, dated April 1, 1991.

The licensee informed the NRC Operations Center of this event via the ENS at about 11:00 p.m. on April 1, 1991.

(5) Insufficient Weld Buildup

In addition to liquid penetrant examinations performed on the thirteen vent and drain line connections in the reactor recirculation system discussed above in sub-paragraph (3), the licensee performed a weld-profile examination. The required weld profile was detailed on Pullman Power Product (pipe installer) isometric drawing B33-5026, Detail B. In order to

accommodate analyzed "faulted-load" conditions (i.e. pressure pulse from high energy line break inside biological shield wall), the original 3/4 inch schedule 40 pipe between the reactor recirculation pipe coupling and the first root valve (a distance of about 1 inch) was to be "built-up" by weld material. The additional weld material was used to increase the wall thickness of the one-inch long pipe material from the schedule 40 to about that of schedule 80. This added wall thickness would then accommodate the analyzed "faulted-load."

The as-found weld profile was not in accordance with the required design for all 14 vent and drain connections. The fourteen welded connections were found to have been made using standard fillet weld details (i.e. 2:1 unequal leg) for both the reactor recirculation pipe coupling socket and the first root valve socket.

The licensee stated that the weld detail was provided by Engineering Change Notice (ECN) 1B33-502 dated September 5, 1984. Further, the licensee stated that the detail used (weld build-up) was unique to the 14 (13 inspected plus the failed connection discussed above in sub-paragraph (3)) unsupported vent and drain connections in the reactor recirculation system. Corrective action for the insufficient weld material was to repair the vent and drain connections by adding weld material.

The licensee researched construction records to determine the cause for the weld profile deficiency. Records showed that the weld detail was properly translated to the installer's field drawings. However, a review of the "field process sheets," "field weld process sheet," examination records, and the required visual examination procedure did not disclose the root cause for the as-found discrepancy. The licensee speculated that the unique weld detail was overlooked when the actual field welds were performed. The inspectors noted that the 14 connections were made by several different welders and inspected by several different inspectors. The only obvious common link between the work performed was the location in the drywell. Since all the welds were made at about the same floor elevation, the weld detail was most likely maintained in the field at the job supervisor's desk. A typical 3/4 inch "socket weld" would have resulted in the weld profile found. Failure of the licensee to assure the weld details described in ECN 1B33-502 were properly controlled is a Violation of 10 CFR 50, Appendix B, Criterion IX, "Control of Special Processes" (NCV 440/91004-02(DRP)). Since this violation met the criteria specified in 10 CFR Part 2, Appendix C, Section V.G., a Notice of Violation was not issued; and, this issue is considered closed.

No deviations were identified. One violation for which a Notice of Violation was not issued was identified.

9. Allegation Review

(Closed) Allegation (RIII-90-A-0118): Licensee management allegedly provided misleading information in Licensee Event Report (LER) 50-440/90020, Dated September 28, 1990.

CONCERNS:

Concern 1: Licensee management provided misleading information in LER 90020 since licensee management, allegedly, were required to report and knowingly failed to report in LER 90020 that a "pinhole" existed in safety-related Control Room Emergency Ventilation System (M25/M26) ducting previous to the hole reported in that LER.

Concern 2: Licensee management ordered that grinding be accomplished in violation of safety-related procedures in that, allegedly, licensee management had ordered individuals to grind on ducting contrary to Work Order (WO) 90-4066 (which allegedly prohibited grinding on the ducting) and contrary to Maintenance Administrative Procedure (MAP)-203, "Conduct of Work," which, allegedly, prohibited grinding on safety-related ducting unless specifically authorized by a safety-related procedure.

Concern 3: The metal patch used to repair the ventilation ductwork reported in LER 90020 was not obtained from the plate specified in Work Order 90-4066.

Concern 4: In order to pass surveillance leak tests, it was routine practice to apply a Dow Corning "grease" on the neoprene portions of louvers of dampers.

INSPECTORS REVIEW:

Concern 1: The inspectors found no basis to conclude that licensee management provided misleading information in LER 90020.

The inspectors determined that the licensee was not required to report in LER 90020 the existence of a "pinhole" that was present in the ducting prior to the rectangular hole reported in that LER. On January 15, 1991, the licensee's system engineer confirmed to the inspectors that a pinhole in the ducting (about .75 square inches) had existed previous to the rectangular hole reported in the LER. The system engineer stated that the pinhole had not caused the control room emergency ventilation system to be inoperable because of the small size of the pinhole and the low flowrate through the torturous path for air escaping from the pinhole. The system engineer further stated that the torturous path was due to the close proximity of insulation and stitch welding of sheet metal to a companion angle flange. The inspectors were not able to examine the pinhole or the

torturous path since that entire section of ducting had been replaced. The inspectors concurred that a pinhole about .75 square inches in the ducting would not have rendered the control room emergency ventilation system inoperable; therefore, there was no regulatory requirement to report the pinhole and licensee management did not omit necessary information in LER 90020.

Concern 2: The inspectors reviewed the applicable procedures, Work Order 90-4066, and Maintenance Administrative Procedure (MAP)-203, "Conduct of Maintenance," and found that neither procedure prohibited grinding on the ducting. The inspectors did not substantiate that licensee management had directed that grinding be accomplished which violated safety-related procedures.

Concern 3: The inspectors substantiated the concern that the metal patch used to repair the rectangular ducthole reported in LER 90020 was not from the plate specified, and documented, in Work Order (WO) 90-4066; however, the inspectors found no evidence of intentional misrepresentation or a significant safety concern.

On December 5, 1990, the licensee initiated Condition Report (CR)-90-433 which documented that a piece of 12-gage galvanized sheet metal from the shop (believed to be ASTM 526 or 527) was used to repair the ducting instead of the material specified in WO 90-4066 and that closed WO 90-4066 did not reflect that the shop metal was used. On January 16, 1991, when contacted by the inspectors, the Maintenance Supervisor that had supervised the ducthole repair stated that on September 1, 1990, at about 1 a.m., he had directed that the repair patch be cut out of the shop metal because the requisitioned material was taking too long to arrive from the warehouse and all shop material was bought with safety-related certifications anyway. He stated that he had been told that if he did not get the hole patched by 4 a.m., then the plant would have to be shutdown.

The Maintenance Supervisor stated that he had not finished the documentation for the Work Order on his shift and that he "forgot" to inform the Maintenance Supervisor for the next shift that the material specified in the work order had not been used. The next shifts' Maintenance Supervisor completed the work order documentation as if the material specified had been used.

Field Clarification Request (FCR) 14766 was written on December 7, 1990, and requested licensee engineering personnel to evaluate whether the shop metal used on September 1, 1990, to repair the ducthole was adequate and whether the control room emergency ventilation system had been correctly declared operable. The licensee's

Engineering response to FCR 14766 stated that the 12-gage galvanized shop metal was adequate to repair the rectangular ducthole because the design specification for the safety-related ducting of the control room emergency ventilation system (Spec 2050 - 4:01.1) only required 18-gage and the use of the thicker and stronger 12-gage galvanized shop metal was conservative. Engineering personnel also performed a visual inspection of a piece of the shop metal used in the repair and concluded that it had more than adequate structural integrity for use in the repair.

Licensee CR 90-433 documented the above inadequate communication and the following corrective actions: Work Order 90-4708 replaced the patch with material of the appropriate certification, Field Clarification Request (FCR) 14766 documented the engineering evaluation that the incorrect material repair was adequate, and the Maintenance Supervisors were trained on the necessity of proper shift turnovers and proper work order documentation. This failure to repair the safety-related ducting with the material specified and documented in Work Order 90-4066 is a Violation (NCV 440/91004-03(DRP)) of 10 CFR 50, Appendix B, Criterion V. Based on the licensee's corrective actions and the minimal safety significance of this licensee-identified event, the NRC is exercising its discretion under 10 CFR 50 Part 2, Appendix C, Section V.G, and is not issuing a Notice of Violation.

Concern 4: At the end of this inspection report period, the surveillance instruction (SVI) that leak tested six ventilation dampers was being separated into two SVIs, one SVI for each train of the control room emergency ventilation system. SVI-M25-T1258A, "Control Room HVAC Air Intake and Exhaust Dampers Leak Test - Train A," Revision 0, was effective March 14, 1991. The system engineer stated that the SVI for the "B" train would be symmetric. Technical Specification 4.7.2.e.4 required that the SVIs be performed "each Refuel Outage (every 18 months)."

The "A" train SVI required that the "as-found" leak rate be determined, that the dampers then be inspected for excessive dirt loading, and that the dampers be cleaned, if necessary, in accordance with the vendor recommendations which included recommendations to maintain lubrication of the damper seals. The vendor manual (File 282, "Pacific Air Products") for dampers: M25-F10A/B, M25-F020A/B, and M25-130A/B recommended that the damper seals be cleaned by frequently removing dirty grease and reapplying a clean coat of Dow Corning 7 or 111 grease.

CONCLUSION:

Concern 1: The concern that licensee management included misleading information in Licensee Event Report 90020 was not substantiated. The inspectors determined that the licensee was not required to report the pinhole that existed in the control room emergency ventilation system ducting previous to the hole reported in LER 90020.

Concern 2: The concern that licensee management ordered workers to violate safety-related procedures was not substantiated.

Concern 3: The concern that the metal plate used to patch the rectangular ducthole was not the material specified in the work order was substantiated; however, the inspectors concluded that the installation of the incorrect metal plate had not degraded the control room emergency ventilation system. No misleading information was in LER 90020. Subsequently, the ducthole was repaired correctly.

Concern 4: The inspectors found no indication of improper usage of grease on dampers. However, the inspectors noted that the governing surveillance instruction did not provide explicit acceptance criteria for the lubricant inspection performed every 18 months. The licensee was addressing the inspectors concern at the close of this report period.

No deviations were identified. One violation was identified for which a Notice of Violation will not be issued.

10. Items For Which A "Notice Of Violation" Will Not Be Issued

The NRC uses the Notice of Violation as a standard method for formalizing the existence of a violation of a legally binding requirement. However, because the NRC wants to encourage and support licensee initiative in the self-identification and correction of problems, the NRC will not generally issue a Notice of Violation for an issue that meets the tests of 10 CFR 2, Appendix C, Section V.G.1. These tests are: 1) the issue was identified by the licensee; 2) the issue would be categorized as Severity Level IV or V violation; 3) the issue was reported to the NRC, if required; 4) the issue will be corrected, including measures to prevent recurrence, within a reasonable time period; and 5) it was not an issue that could reasonably be expected to have been prevented by the licensee's corrective action for a previous violation. Issues involving the failure to meet regulatory requirements, identified during the inspection, for which a Notice of Violation will not be issued are discussed in paragraphs 4, 8.b.(5), and 9.

11. Exit Interviews

The inspectors met with the licensee representatives denoted in Paragraph 1 throughout the inspection period and on April 17, 1991. The inspector summarized the scope and results of the inspection and

discu e the likely content of the inspection report. The licensee did not in e that any of the information disclosed during the inspection could be considered proprietary in nature.