U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-440/92017(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: July 13 - August 31, 1992

Inspectors: A. Vegel
P. Hiland
J. Hopkins

Approved By: Philip Brochman for 9/17/92
R. D. Lanksbury, Chief Date
Reactor Projects
Section 3B

Inspection Summary

Inspection on July 13 through August 31, 1992 (Report No. 50-440/92017(DRP))
Areas Inspected: Routine unannounced safety inspection by resident and region based inspectors of previously identified items, licensee event report follow-up, surveillance observations, maintenance observations, operational safety verification, followup of concerns, temporary modifications, plant record verification (TI 2515/115), licensee evaluation of changes to environs (TI 2515/112), reliable decay heat removal during outages (TI 2515/113), event follow-up, evaluation of licensee self-assessment capabilities, and management meetings.
Results: Of the 12 areas inspected, no violations were identified in 11 areas; one non-cited violation (NCV) was identified in the area of licensee event report follow-up (failure to verify containment temperature and relative humidity - paragraph 3.b).

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

Plant Operations

- The reactor plant was operated at or near full power continuously during this report period.
- Routine activities performed by plant operations were conducted in accordance with existing procedures.

Maintenance/Surveillance

- The quality of observed maintenance and surveillance activities was good.

Engineering and Technical Support

- Good engineering and technical support of daily plant activities and prompt evaluation of previously identified problems with compression fittings was noted.

Safety Assessment and Quality Verification

- The quality of reviewed event reports was acceptable.
- The On-Site Review Committee was evaluated as effective.
- Further evaluation of individuals not performing required inspections will be tracked as IFI 50-440/92017-01(DRP).

DETAILS

1. Persons Contacted

a. Cleveland Electric Illuminating Company

M. Lyster, Vice President - Nuclear
#*R. Stratman, General Manager, Perry Nuclear Power Plant (PNPP)
*K. Donovan, Manager, Licensing and Compliance
M. Gmyrek, Operations Manager, PNPP
S. Kensicki, Director, Perry Nuclear Engineering Department (PNED)
F. Stead, Director, Perry Nuclear Support Department (PNSD)
*H. Hegrat, Compliance Engineer, PNSD
E. Riley, Director, Perry Nuclear Assurance Department (PNAD)
#*V. Concel, Manager, Technical Section, PNED
*D. Conran, Compliance Engineer, PNSD
W. Coleman, Manager, Quality Assurance Section
P. Volza, Manager, Radiation Protection Section
D. Cobb, Superintendent, Plant Operations, PNPP
F. Von Ahn, Mechanical Engineering Unit Lead, PNED
W. Kanda, Manager, Electrical Design Section, PNED
R. Gaston, Compliance Engineer, PNSD
*M. Hayner, Supervisor, Licensing Engineer PNSD
*J. Messina, Shift Supervisor, PNPP
B. Beyer, Director, Perry Administrative Services Department, PASD
W. Wright, Manager, Instrumentation and Controls Section, PNPP
K. Pech, Manager, Outage Planning Section, PNPP
M. Cohen, Manager, Maintenance Section, PNPP
B. Walrath, Manager, Performance Engineering Section PNED

b. U. S. Nuclear Regulatory Commission

E. Greenman, Director, Division of Reactor Projects (DRP), RIII
R. Greger, Chief, Branch 3, DRP, RIII
J. Hall, Senior Project Manager, Office of Nuclear Reactor Regulation
P. Hiland, Senior Resident Inspector, RIII
#*A. Vegel, Resident Inspector, RIII

# Denotes those attending the management meeting held on July 31, 1992.
* Denotes those attending the exit meeting held on August 31, 1992.

2. Licensee Action on Previous Inspection Findings (92701)

a. (Closed) Open Item (50-440/90002-02(DRP)): Human Factors Recommendations Following Review of Reactor Scram. On January 7, 1990, an automatic reactor scram occurred due to a personnel error. The wrong fuses were removed during a preventive maintenance task causing a loss of power to the feedwater control system. Seven recommendations were provided to the licensee for their consideration at the conclusion of the scram event review previously documented in Inspection Report 50-440/90002, Paragraph 8.b.(2), dated March 20, 1990.

During this report period, the inspectors reviewed the status of this item with the licensee. The inspectors noted that the licensee had evaluated the seven recommendations and incorporated several. Of note was the separation of Unit 1 and Unit 2 plant electrical instructions; a maximum time limit of 2 minutes was assigned for parallel operation of the 480 volt supply transformers; and the implementing checklist, in Attachment 34 to System Operating Instruction (SOI) 1R10(LV), required a pre-evolution brief and verification that the tie breaker was closed before opening the supply breaker. Based on the inspectors review of licensee corrective actions described in LER 50-440/90001 and the licensee review of the subject recommendations, this item is closed.

b. (Closed) Inspection Followup Item (50-440/92003-01(DRP)): Compression Tube Fittings. As documented in Inspection Report 50-440/92009, dated June 18, 1992, the inspectors had previously noted the licensee's investigation and extensive corrective actions for the problems with compression fittings, including mixed components and the "pullup" problem. This item had remained open, pending the inspectors follow-up review of the licensee's long term corrective action plans, including the licensee's evaluation of NRC Information Notice (IN) 92-15.

The inspectors reviewed the licensee's assessment of NRC IN 92-15, "Failure of Primary System Compression Fittings." The licensee addressed the IN by reviewing previously identified problems with compression fittings, evaluating their training program, and developing a long term corrective action plan. To identify deficiencies with compression fittings described in IN 92-15, the licensee incorporated into the quarterly outage schedules an inspection program to check tubing connections not included in previous inspections. During the next 36-month period, whenever systems or equipment were to be taken out of service for maintenance or calibration, the adjacent readily accessible compression tube fittings were to be inspected for deficiencies. The 36-month period for this program was chosen because safety-related instruments were calibrated on an 18-month cycle and the nonsafety-related instruments were calibrated on a 36-month cycle. In addition to the inspection program, the maintenance training program was enhanced to address the "pullup" criteria specific to different compression fitting manufacturer's requirements.

The inspectors concluded that the licensee's initiatives appeared adequate in scope to identify and correct compression fitting problems. The effectiveness of the licensee's program will be evaluated in future routine inspections of plant systems and maintenance activities. This item is closed.

3. Licensee Event Report (LER) Follow-up (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 50-440/89006-01: During the first Perry refueling outage, Type B and C leakage rates exceeded TS limits. In addition, required testing on two residual heat removal (RHR) system flanges was not performed.

Licensee Investigation of Root Cause and Corrective Actions

Root Cause

As stated in the LER, many factors contributed to the identified leak rate test failures. Examples included: disc-to-seat misalignment, oxide buildup, popped seat defects, out of round wear, and small discontinuities. The failure to leak test the two RHR flanges was a program deficiency.

Corrective Action

The identified leakage pathways (valves) were repaired by mechanical means or replaced. The leak rate test program was revised to include the two RHR flanges.

Inspectors Review

The inspectors noted that the licensee reported similar leak rate test failures in 1990 (reference LER 50-440/90025). As a result of those failures, the licensee committed to modify main steam isolation valves (MSIVs) to incorporate the latest technical improvements. In 1992, six of the eight MSIVs were modified as discussed in LER 50-440/92006. The inspectors noted that Design Change Package (DCP) 89-00224 was implemented adding components required to test the two RHR flanges. This item is closed.

b. (Closed) LER 50-440/90018-00: On August 17, 1990, containment average temperature and/or relative humidity had not been verified to be within TS limits for a period of approximately 6 days. For undetermined reasons, process computer data acquisition for containment humidity had been secured on August 12, 1990; however, the failure of the computer display to update was not recognized by control room personnel. The process computer point was immediately restored to normal and an instrumentation and control (I&C) technician was dispatched to measure containment relative humidity using a psychrometer to confirm TS compliance.

Licensee's Investigation of Root Cause and Corrective Actions

Root Cause

The licensee determined the root cause of the event was a programmatic weakness in the use of the process computer system information to determine TS compliance. Although instructions specifically require the process computer to be used to obtain containment humidity data, formal training and procedural guidance was inadequate to ensure effective and consistent use by the control room operators.

Corrective Action

To prevent recurrence, initial and requalification training programs for licensed operators were enhanced to include additional training on the process computer. Additionally, control room administrative guidance was developed to assist the operators in evaluating process computer data displays.

Inspectors Review

The inspectors reviewed applicable licensee documentation and noted that all corrective action commitments were completed. Licensee corrective actions appeared adequate to prevent recurrence. The licensee's failure to verify containment average temperature and/or relative humidity at least once every 24 hours between August 12 and August 17, 1990, was a violation of TS 4.6.5.2. This violation was not cited because the licensee's efforts in identifying and correcting the violation met the criteria specified in section VII.B of the "General Statement of Policy and Procedure for NRC Enforcement Actions," (Enforcement Policy, 10 CFR Part 2, Appendix C (1992)). This item is closed.

c. (Closed) LER 50-440/91010-00: Faulty Electrical Contactor results in a Reactor Recirculation (RR) Pump Trip. During a plant shutdown to investigate an increase in drywell unidentified leakage, RR pump "A" failed to downshift to slow speed and resulted in a pump trip. In accordance with TS, the plant was shut down.

Licensee's Investigation of Root Cause and Corrective Action

Root Cause

As stated in the LER, the failure of an electrical contactor to close its contacts was identified as the cause for the RR pump trip. The failure could not be duplicated during post event testing and was believed to be caused by dirt and/or dust preventing correct contact operation.

Corrective Action

The electrical contactor was replaced in RR pump "A". In addition, a repetitive maintenance task was implemented to clean the electrical cubicles for both the "A" and "B" RR pumps.

Inspectors Review

Based on the inspectors review of completed corrective action for failure of the RR pump electrical contactor to properly function, this item is closed.

The inspectors noted the cause for the increased drywell leakage (precursor to this event) was previously reviewed by the inspectors in inspection report 50-440/91004, Paragraph 7.b.(3). The inspectors noted that related LER 50-440/91025, concerning weld cracks on the 3/4-inch (1.9 cm) high pressure core spray test return line vent, was still open.

d. (Closed) LER 50-440/91014-00: On July 26, 1991, a reactor water cleanup (RWCU) system containment isolation occurred due to a high differential temperature signal in the RWCU pump room. During decontamination activity, a leak detection system temperature element located in the RWCU "A" pump room area was sprayed with water resulting in the high differential temperature signal system isolation and pump trip. After verifying that no steam leak existed, the RWCU "B" pump was returned to service.

Licensee's Investigation of Root Cause and Corrective Actions

Root Cause

The licensee determined the cause of the event was personnel error (inadequate training and/or knowledge). The health physics technician who directed the decontamination process and the maintenance personnel performing the decontamination were unaware of the consequence of wetting this instrumentation. There were no signs in the area to communicate the sensitive nature of this instrumentation to personnel performing the decontamination.

Corrective Action

To prevent recurrence, the licensee took the following corrective actions. The event was discussed during continuing training for health physics personnel and during initial decontamination techniques training for plant helpers, health physics personnel, and contractors used to supplement decontamination activities in the plant. Additionally, the event was discussed with licensed operators as part of the requalification training program. Signs identifying differential temperature elements were permanently posted designating them as "Temperature Sensitive Elements."

Inspectors Review

The inspectors reviewed 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 event recurrence. This item is closed.

e. Additional LERs Reviewed

In addition to the above LERs, the inspectors reviewed the following reports to determine that the reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence was completed or initiated.

LER	TITLE
440/89010-01	Heat Damaged Cables
440/89011-00	Auto Start of Emergency Service Water Pump
440/89030-01	Control Rod Untrippable
440/90001-00	Reactor Scram due to Loss of Feedwater
440/90009-00	Loss of Containment Spray and Pool Cooling
440/90016-00	HPCS Suction Valve Shift
440/90021-01	MSIV Fail to Close
440/90025-02	Leak Rate Test Failures (followup in LER 50-440/92006)
440/90026-01	Failed Bypass Leak Rate Test
440/90030-00	Surveillance Prior to Fuel Move

No deviations were identified; however, one non-cited violation (NCV) was identified.

4. Followup of Concerns (62703)

a. (CLOSED) Supervisory Direction Concerns (RIII-92-A-0042): The senior resident inspector, with assistance from a region III health physics inspector, reviewed the following concerns regarding direction given to site employees during the Spring 1992 refueling outage.

Concern 1. Employees are threatened to not report violations. Tradesmen have been requested to report no wrongdoing to their management.

Concern 2. Certifications are signed when work has not been completed or inspected by qualified personnel.

Concern 3. Employees are told dosimeters are not required while working in containment.

Discussion:

For the purpose of evaluating Concern 1 and 3, the senior resident inspector and a Region III health physics inspector conducted interviews of craft personnel during the Spring 1992 refueling outage. The purpose of these interviews was to determine if a clear understanding of available methods to report known violations or equipment problems was apparent in the work groups. In addition, interviews were conducted at job locations within the containment and included discussions on required dosimetry. About 20 persons (plant employees and contractors) were interviewed. Personnel interviewed were aware of requirements for wearing dosimetry inside containment and were unaware of any direction to the contrary. The inspectors noted that the personnel interviewed were provided appropriate dosimetry. Persons interviewed exhibited an adequate knowledge on available methods to report violations or equipment problems and were unaware of any direction or threat to the contrary.

During this report period, the inspector reviewed plant records concerning the improper use of dosimetry at the Perry site. Between February and July 1992, four "Radiological Occurrence Reports" were initiated that documented 22 instances where site personnel did not use proper dosimetry while working within the radiological restricted area (RRA) including containment. None of the 22 instances resulted from improper direction to the involved individuals. The failure of the individuals to recognize their lack of proper dosimetry prior to entering the RRA was the dominant root cause for these events. Inadequate training and overall poor planning of changes to a dosimetry program procedure were contributing causes. Assigned doses were less than 100 millirem [1 mSv] for each of the events, and were based on secondary dosimeter data or time and motion evaluations using survey data. Dose assignments appeared reasonable. Corrective actions for the initial events may have been too narrowly focused, but actions taken after the subsequent events appeared wide ranging.

For the purpose of evaluating Concern 2, the senior resident inspector reviewed 82 safety-related work orders completed during the 1992 refueling outage. Work orders reviewed contained one or more of the following: multiple Hold and Witness points; American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME) Form NR-1, "Owners Report for Repair or Replacement"; ASME Form NV-1, "Manufactures Data Report for Safety and Safety Relief

Valves"; ASME Form N-2, "Certificate Holders Data Report for Identical Nuclear Parts and Appurtenances"; or inspection records such as visual (VT) and leak rate test results. In general, for the work orders reviewed, Hold and/or Witness points were signed in accordance with the requirements of the work package. Some Witness points were not performed; however, an appropriate explanation or revision to the work package was provided. All certifications on required ASME forms and inspection reports were documented prior to final closure review. The inspector noted that, as part of the work order package, a final Quality Assurance review was performed. Between April and July 1992, six work orders were identified by the licensee where failure to obtain proper signatures (4 Hold points and 5 Witness points) occurred. The inspector reviewed the associated licensee "Action Request" documenting those missed Hold and Witness points and noted appropriate corrective action was planned or implemented.

Conclusion:

Based on the inspectors interviews conducted during the 1992 refueling outage, Concern 1 was not substantiated.

Based on the inspectors review of work orders and licensee corrective action documents, Concern 2 was partly substantiated. As noted above, six work orders were identified by the licensee where Hold or Witness points were missed. That identification and associated corrective action was part of the normal work order review process (i.e. prior to final closure). About 1,600 work orders were performed during the 1992 refueling outage.

Based on the interviews of site personnel conducted during the refueling outage and the inspector's review of RORs initiated between April and July 1992, Concern 3 was not substantiated.

b. (Closed) Control Rod Blade Movement Concern (RIII-92-A-058):

During this report period, the inspectors reviewed existing licensee controls over the movement of control rods in the refueling and the fuel handling pools. The purpose of this review was to evaluate whether a potential generic concern was a problem at the Perry plant.

Concern: Inadequate controls to restrict upward travel of control rod blades when using the frame mounted hoist on the refueling bridge and fuel handling bridge.

Discussion: The inspectors reviewed the Perry updated safety analysis report (USAR) Section 9.1 and noted the minimum water coverage above control rod blades was stated to be 6 feet 7 inches (2 m) at the spent control rod storage location in the fuel building storage pool. Movement of irradiated control rods to the storage location was controlled by system operating instruction (SOI) F11/F15, "Fuel Handling, Refueling and Auxiliary Platforms," refueling instruction (FTI) E02, "Installation and Removal of

Control Rod and/or a Guide Tube Seal," and fuel accountability instruction (FTI) D06, "Preparation of Fuel Movement Checklist."

At Perry, hoists used to move control rods included a frame mounted auxiliary hoist on the refueling platform (F15) and a monorail hoist on both the refueling platform (F15) and the fuel handling platform (F11). All three hoists incorporate the use of geared limit switches and mechanical stops to limit the upward travel. The system operating instructions for the hoists on both the refueling platform (containment) and fuel handling platform (fuel building) were contained in SOI-F11/F15. Specific instructions for control rod grapple installation onto the hoists was contained in FTI-E02. The normal upward travel on the hoists was limited by the "Up Travel Stop" electrical limit switches (2 switches per hoist). As a backup to the electrical limit switches, mechanical stops (cable clamps) were installed on the hoist cable about 2 inches (5.08 cm) below the upper cable guide when in the "Up Travel Stop" position. Failure of the electrical limit switches would result in the cable clamps contacting the cable guides preventing further upward travel. Instructions for the installation and adjustment of the mechanical stops were contained in FTI-E02, Paragraph 5.1.12 and 5.2.13.

The inspectors noted, through review of the associated refueling and fuel handling platform vendor equipment manuals (licensee Nos. 128G and 146G), the use of mechanical stops ("adjustable plate") was described. Specifically, interlock switch "S8" was actuated by the described mechanical stop.

Just prior to the spring 1992 refueling outage, the electrical monorail interlock for the fuel handling platform was adjusted in accordance with setpoint change request (SCR) 91-106R. The inspectors reviewed the implementing work order (WO 92-625) and noted that change resulted in a control rod submergence of 6 feet 10 inches (2.08 m). With the mechanical stop installed at 2 inches from the electrical setpoint, that would provide 6 feet 8 inches (2.03 m) of submergence for an irradiated control rod consistent with the minimum depth stated in the USAR.

Conclusion: Based on the review of SOIs, vendor technical manuals, and WOs performed prior to irradiated control rod movement evolutions during the spring 1992 refueling outage, the inspectors concluded that adequate controls were in place to restrict upward travel of control rod blades. This concern was not substantiated.

No violations or deviations were identified.

5. 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

SVI-R42-T5520	125 V Battery Category B limits, Thermal Corrosion and Electrolyte Temperature Check (Unit 1 Division II)
ICI-C-C34-1	Loop Calibration of 1C34-N002A and 1C34-N002B
SVI-G43-T1305A	Suppression Pool Water Level Channel Calibration for 1G43-N060A

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 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
R86-4255	Division III diesel generator breaker cleaning and service.
R87-3507	Instrument Air Filter Changeout
W092-3057	Clean Instrument Air Receiver Tank
W092-0493	Adjust Limits and Mechanical Stops for OGH41F0360

WO92-1025	Replace pin vent on 1G43N0060A
WO92-2608	Division II Diesel Generator Painting

No violations or deviations were identified.

7. Operational Safety Verification (71707)

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

The inspectors observed plant housekeeping, general plant cleanliness conditions, and verified implementation of radiation protection controls.

a. Control of Annunciators

The inspectors reviewed the licensee's control of disabled or partially disabled control room annunciators. As a result, the inspectors identified multiple examples of annunciators being labeled as Not in Service (NIS) stickers still in place even though the corrective action (i.e. repairs or modifications) had been completed.

The inspectors reviewed Perry administrative procedure (PAP) 1404, "Miscellaneous Tagging," which defined and implemented tagging procedures for various tags including NIS stickers. Not in Service stickers were used to inform the control room operators when the annunciator or other indicator was out of service or out of calibration, such that the associated control room readout was incorrect or inoperable. The following annunciators still had NIS stickers installed though the specific deficiency identified on the NIS sticker had been corrected.

Not in Service Sticker	Affected Annunciator
1-91-174	Steam tunnel Leak detection ambient temperature P-632
1-90-023	RHR B 005
1-90-057	Valve stem leak-off Temp P-865

1-89-022	Floor Drain Collector Tank B Conducting High
1-89-023	Waste Collector Tank B Conductivity High
1-89-025	Concentrate Waste Heat Tracing Temperature High/Low
1-88-068	Hot water heating trouble

The inspectors identified these inaccurate NIS stickers to the licensee. The inspectors were concerned that as a result of the NIS stickers still being in place, the control room operators were not fully cognizant of the actual status of the annunciators. The licensee took immediate corrective action to review the NIS index and remove the stickers that were no longer applicable. The licensee determined the cause for the NIS sticker discrepancies was that no administrative means were in place to ensure that once a WO correcting an annunciator problem was completed, the associated NIS sticker was also cleared. To prevent recurrence, the licensee initiated action to modify the work request process so that NIS stickers were included in work package closure. In addition, a quarterly review of the NIS sticker index was to be conducted by a senior reactor operator to verify the status of the NIS stickers. The inspectors will review the effectiveness of the licensee's corrective actions during future routine inspection of control room activities.

No violations or deviations were identified.

8. Temporary Modifications (37700)

During this inspection period the inspectors conducted an evaluation of the licensee's control of temporary modifications. The inspectors reviewed the licensee's temporary modification control procedure (PAP-1402), reviewed the Mechanical Foreign Items and/or Lifted Leads, Jumpers and Electrical Devices (MFI/LLJED) index, conducted a sample field verification of tags, and discussed implementation of the program with the responsible licensee management. The inspectors reviewed the MFI/LLJED index to verify that 10 CFR 50.59 evaluations were conducted for each entry and that control room drawings were annotated in accordarce with procedural requirements. As a result of these reviews, the inspectors did not identify any deficiencies with the licensee's implementation of the temporary modification procedure. Though the licensee's program to control temporary modifications was apparently effective in documenting the MFI/LLJEDs installed in the field, the inspectors were concerned with the length of time that some of these temporary modifications were installed prior to a permanent fix being implemented.

The licensee's administrative procedure which implemented the control of the temporary modifications program (PAP-1402), stated that MFI/LLJEDs should normally be installed for no longer than one operating cycle. The procedure further stated that MFI/LLJEDs should not be permitted as

interim alternatives to permanent design changes absolutely needed to support continued plant operation. The inspectors identified multiple MFI/LLJEDs that had been installed in the plant for several operating cycles. At least 33 of the 70 MFI/LLJED's alterations presently installed in the field have been in place for more than 1 operating cycle. As stated in the licensee's temporary modification procedure, the control of the number and duration of MFI/LLJED's is necessary in order to ensure that systems and components perform as designed and that distractions to shift operating personnel are minimized. Though the inspectors recognize that temporary modifications may be required to support continued plant operation, long duration use can detract from the ability of plant operators to determine and maintain control over plant status. The inspectors concluded that the administrative procedure in place was inadequate to ensure that temporary modifications were being effectively evaluated in a timely manner to expedite modifications as required.

The licensee was aware of the problems concerning the duration of temporary modifications and was in the process of revising the temporary modifications procedure to address the issue. Specific planned changes to the procedure included the required approval for extended installation of a temporary modification by the plant manager or generation of a drawing change notice. In addition, the responsibilities of various licensee organizations was clarified with respect to tracking and scheduling the removal of MFI/LLJEDs. To improve the tracking of MFI/LLJEDs, the licensee recently enhanced the tracking mechanism which lists all the temporary modifications, related DCPs and work orders, and their status. The purpose of this list was to provide licensee management with a more comprehensive listing of the temporary modifications installed and what action is planned to address them.

The inspectors concluded that the licensee has initiated corrective action to improve the temporary modification control process through procedure enhancement. The effectiveness of the licensee corrective actions will be evaluated in future inspection reports.

No violations or deviations were identified.

9. Onsite Followup of Events at Operating Power Reactors (93702)

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 inspector follow-up are provided in Paragraphs a and b below.

a. Reactor Thermal Power Limits Potentially Exceeded

On July 17, 1992, the licensee determined that actual reactor thermal power may have been higher than indicated power by approximately 10 megawatts thermal (MWt). A Heise digital pressure gauge, serial number S7-19060, used for the calibration of the reactor feedwater flow transmitters during the third refueling outage, was determined to be out of calibration (OOC). The licensee immediately reduced reactor power to 99.7 percent and administratively limited power to 3569 MWt, 10 MWt below the license limit. The licensee informed the NRC Operations Center of this event via the ENS at about 5:00 p.m. on July 17, 1992.

On July 20, 1992, the feedwater flow transmitters were recalibrated using surveillance ICI-C-C34-1, "Loop Calibration of 1C34-N002A and 1C34-N002B." The voltage and pressure values on both transmitters and total flow circuits were within tolerance. The licensee reviewed the data and determined that the feedwater flow transmitters were indicating correctly and that the licensed thermal power limit had not been exceeded. The administrative limit on reactor power was removed and power was increased to 100 percent. On July 24 the licensee informed the NRC Operations Center and retracted the July 17 notification.

The licensee initiated condition report (CR)-92-189 to document the results of their investigation into the causes of the event and the corrective actions taken. While reviewing CR-92-189, the inspectors were concerned about the time delay between the initial identification of the OOC and the determination that licensed thermal power may have been exceeded. An evaluation of the OOC report was due 30 days from the discovery date. The feedwater control system engineer (RSE) identified the potential for exceeding thermal power limits on July 17.

Although the evaluation of the OOC report was completed within the allotted 30 day period, the inspectors were concerned with the apparently excessive time period between the time the RSE received the report (July 9) until the evaluation was completed (July 17). The inspectors discussed their concern with the RSE and licensee management. Based on these discussions and review of the OOC reporting process, the inspectors concluded that the RSE reviewed the report in a timely manner and that the program to evaluate OOC reports apparently functioned as intended. Taking into consideration the RSE's prioritization of tasks assigned and the extent of review required to determine the impact of an OOC instrument on system performance, the inspectors concluded that the OOC report was reviewed in a timely manner.

b. Loss of Emergency Communications Capability

On August 1, 1992, at 9:05 a.m., while in Operational Condition 1, POWER OPERATIONS, the plant experienced a significant loss of telephone communications due to a loss of power to a telephone fiber optics cabinet. At approximately 9:10 a.m., power was restored to the affected cabinet and full communications

capabilities were re-established. An Unusual Event was entered and simultaneously terminated at 9:40 a.m., in accordance with plant emergency plan implementing instructions (EPI-A1:1.1.1 (page 59), "Significant Loss of Communications Capability").

The event occurred during the implementation of a scheduled design change to the main power distribution panel which normally supplied power to two telephone fiber optics cabinets. These fiber optics cabinets connect the plant private branch exchange (PBX) switches and dedicated lines to the local Perry telephone exchange. Power was lost to one of these fiber optics cabinets due to a loose connection on a temporary power jumper. Without the temporary power available, the cabinet lost all power once the main distribution panel was deenergized in support of the design change. As a result, the dedicated state and/or county notification circuit, NRC emergency notification system (ENS) circuit, private offsite lines, and both plant PBX switches were lost. The plant Off-Premises Exchange (OPX) circuit remained in service and provided the control room with limited offsite calling capability. The OPX phones utilized a microwave link and were independent of the fiber optic cabinets. Communications capabilities were restored once the temporary power jumper was reconnected. Subsequently, all affected phone lines were tested satisfactorily. The licensee initiated CR 92-196 to document the results of their investigation into the cause of this event and the corrective actions taken.

The licensee informed the NRC Operations Center of this event via the PBX phone system at about 10:08 a.m. on August 1.

No deviations or violations were identified.

10. Reliable Decay Heat Removal During Outages (TI 2515/113)

An inspection of the licensee's plan to ensure reliable decay heat removal during refueling outage 3 (RF-3) was documented in inspection report 50-440/92003(DRP). The inspectors conducted a followup review using guidance contained in Temporary Instruction (TI) 2515/113, "Reliable Decay Heat Removal During Outages," to assess the licensee's continued implementation of the program.

The inspectors observed the licensee implementing its programs for ensuring reliable decay heat removal during daily outage planning meetings, control room pre-shift briefings, and status meetings. The operable shutdown cooling system(s), AC power sources, ECCS train(s) available, alternate decay heat removal methods, and higher risk activities were all identified on the refueling plan of the day and discussed at the meetings.

The replacement of control rod drive mechanisms (CRDMs) and control rod blades (CRBs) was identified as a higher risk activity with the potential to drain the reactor vessel. To minimize the potential for working on the same fuel cell at the same time, CRDM and CRB replacements were scheduled in separate core quadrants. A review of the work plan determined that adjacent fuel cells were scheduled for

concurrent work even though they were in different core quadrants. The work plan was subsequently revised to prevent this situation.

A weakness was identified concerning the control room operators' ability to use the decay heat removal capacity curves. The curves were developed to assist the operators in determining if an alternate shutdown cooling system was capable of removing the decay heat being generated. The licensee determined that poor labeling and instructions were the root causes. Shift training was conducted as short term corrective action. The labels and instructions were being revised for RF-4.

The inspectors concluded that the licensee implementation was conservative and effective at minimizing risk to the core. Based on this review, the inspectors have no further concerns and TI 2515/113 is considered closed.

No violations or deviations were identified.

11. Licensee Evaluations of Changes to the Environs Around Licensed Reactor Facilities (TI 2515/112)

A review of the licensee's existing programs to evaluate changes to the environs surrounding the Perry plant was conducted during this report period. The purpose of this review was to determine if the licensee was updating the Perry safety analysis report to reflect changes in the licensing basis.

Licensee Administrative Controls

Perry Administrative Procedure (PAP) 0520, "Changes to Licensing Documents and License Amendments," delineated the responsibilities for changes to the updated safety analysis report (USAR) in accordance with 10 CFR 50.71. As stated in PAP-0520, the Perry USAR was to be revised at least annually on or before March 18 of each year. The Manager, Licensing and Compliance (LCS), was responsible for these revisions. Based on discussions with the cognizant LCS supervisor, the inspectors noted that proposed revisions were solicited from all site organizations on a yearly basis. The solicited inputs were processed as a "PNPP Change Request" in accordance with PAP-0520.

Inspectors Review

The inspectors noted that revisions to USAR Section 2.0 had been made in each of the four revisions issued since the Perry plant was licensed. The inspectors reviewed historical USAR change requests, provided by the licensee, as examples of revisions reflecting an awareness of changing environs. In addition, the inspectors discussed with cognizant emergency planning personnel, the results of their periodic "Evacuation Time Estimate" and the feedback provided on changes to the population.

The inspectors noted that the USAR yearly update required the responsible manager to solicit input from appropriate site and licensee corporate organizations. No formal requirement existed to pro-actively survey the environs for changes in population distribution or site

proximity hazards. A population survey was performed in 1991 to update the "Evacuation Time Estimate." As noted above, the results of that survey were provided to the manager responsible for the USAR update. The next population survey was tentatively planned for 1995 or 1996. The licensee relied on "self awareness" of changes in site proximity hazards. An example provided by the licensee (CR 89-155) was a revision to the USAR due to gas lines near the site having an increase line pressure of 35 to 60 psi [241 to 414 kPa].

The inspectors concluded that a specific program did not exist that would assure, over the life of the plant, changes to the environs would be identified. Specifically, the lack of a periodic requirement to survey businesses described in Section 2 of the USAR resulted in reliance on "self awareness." The population survey conducted to support emergency planning was provided to the manager responsible for USAR updates; however, that survey was performed on a verbal commitment to local agencies participating in the emergency planning efforts.

The inspectors noted that change requests to the USAR were controlled by an existing administrative procedure which defined the review and approval process. For change requests reviewed, the inspectors noted the bases for the conclusions relative to impact on public health were documented. Based on the inspectors' review, TI 2515/112 is closed.

No violations or deviations were identified.

12. Verification of Plant Records (TI 2515/115)

An inspection, in accordance with TI 2515/115, "Verification of Plant Records," was performed to assess the licensee's actions to assure the accuracy of logs and records. Both safety- and nonsafety-related logs taken by nonlicensed Perry Plant Operators (PPO) from March through July 1992, were randomly selected and evaluated.

Prior to the inspectors review, the licensee conducted a review of PPO logs in response to IN 92-30. Ten sets of rounds from April through June 1992 were reviewed with no discrepancies being identified. In addition, a self-monitoring program, quarterly repetitive task 92-697, was created to verify operator rounds using security door access records. The licensee also took credit for the plant equipment rounds performance checklist review, Operations Administrative Procedure 1702, "Evaluation of Plant Equipment Rounds," section 6.1.3, as another check that ensured the required areas had been entered. The Quality Assurance department conducted an independent review using the guidance in TI 2515/115.

During the inspectors review of plant records, two separate examples where the PPOs logs and the security door access records did not match were identified. The dates of the discrepancies were May 7, 1992, (8:00 pm - 8:00 am shift) and June 25, 1992, (4:00 pm - midnight shift). Both examples were entries into the Unit 1 and 2, Division 1 and 2 cable rooms (638' elevation, Control Complex) for a general inspection. These examples were referred to the licensee to determine if falsification had occurred.

The licensee (Operations Section) conducted an evaluation and confirmed the inspectors findings. Disciplinary action was taken against both operators while the licensee expanded its investigation, due to additional discrepancies for one of the PPOs. The licensee determined that the missed entry for one PPO appeared to be an isolated occurrence. However, the licensee terminated the second PPO when a total of seven examples were identified. During the expanded search, the licensee identified another individual who had not entered the Unit 1 and 2, Division 1 and 2 cable rooms, July 25, 1991 (midnight - 8:00 am shift). The licensee determined that this missed room entry was an isolated occurrence. The PPOs received disciplinary action.

The missed room inspections had minor safety significance because the room was entered several times throughout the shift by other personnel including fire and security watches, there were no TS or safety-related readings, and the room was entered by the PPO on previous and subsequent shifts with no discrepancies identified. The specific written requirement to enter the room was documented in Plant Equipment Rounds (PRI-PER), Attachment 2.

During the licensee's expanded investigation, two areas in the Service Building were identified where general area room inspections were not consistently being performed. Operators inspecting an electrical equipment room on the 605' elevation and the hot machine shop on the 620' elevation would only complete a partial area inspection or view the area through a window. The Operations Section interviewed the operators and determined that these inconsistencies were due to a misunderstanding of PRI-PER requirements for proper general area inspections and not a willful falsification of plant records.

On August 27, 1992, the resident inspectors were informed by the licensee that two additional employees were terminated as a result of a review of fire technician duties. The licensee determined that two fire technicians failed to perform their duties in accordance with periodic test instruction (PTI)-P54-P0044, "Fire Door Verification." During review of security door access records for February 15, 1992, the licensee discovered significant discrepancies between the PTI record sheet and the security doors accessed by the individuals. In the case of one of the fire technician's key card access records, no entries other than entrance and exit from the site were recorded.

Based on the results of this inspection, TI 2515/115 is closed. The three examples of nonlicensed operators not inspecting the Unit 1 and 2, Division 1 and 2 cable rooms and the two fire technicians not performing their required rounds are being tracked as an Inspection Follow-up Item (50-440/92017-01(DRP)).

No violations or deviations were identified.

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

The inspectors observed selected On-site Review Committee meetings to evaluate that organizations's effectiveness. For the meeting attended, the inspectors considered the following attributes: 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 chairman; if the committee was biased toward operation or safety; and, if the committee used design basis, USAR, or vendor technical manuals for their determinations in addition to the Technical Specifications.

In preparation for the attended meeting, the inspectors reviewed draft submittals of items that were submitted for the on-site review committee's 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
92-0101	8/20/92

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.

14. Plant Status Meeting (30702)

NRC management met with licensee management on July 31, 1992, at the Perry Plant. Personnel attending that meeting are designated by (#) in Paragraph 1 of this report. The purpose of the meeting was to discuss the licensee's efforts to reduce personnel errors, licensee's assessment of the third refueling outage, and review of the four plant events documented in special inspection report 50-440/92011, dated July 2, 1992. At the conclusion of the meeting, NRC management acknowledged the licensee's efforts to decrease personnel errors and improve plant performance.

15. Items For Which A "Notice of Violation" Will Not Be Issued

During this inspection, certain activities, as described above in paragraph 3.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.

16. Inspection Follow-up Items

An Inspection Follow-up Item (IFI) is a matter which has been discussed with the licensee, which will be reviewed further by the inspectors, and which involve some action on the part of the NRC or licensee or both. An IFI identified during the inspection is discussed in paragraph 12.

17. Exit Interviews

The inspectors met with the licensee representatives denoted in Paragraph 1 throughout the inspection period and on August 31, 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
A. Januska	07/24/92
G. Nejfelt	08/07/92
R. Westberg	08/21/92