U. S. NUCLEAR REGULATORY COMMISSION

REGION III

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Licensee:	Centerior Service Company
Facility:	Perry Nuclear Power Plant
Location:	P. O. Box 97, A200 Perry, OH 44081
Dates:	October 4 to December 1, 1997
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EXECUTIVE SUMMARY

Perry Nuclear Power Plant
NRC Inspection Report No. 50-440/97016(DRP)

This inspection included a review of aspects of the licensee's operations, maintenance, engineering, and plant support functional areas. The report covers an 8-week period of resident inspection. One violation of NRC requirements was identified.

Operations

- Thorough preparations were made prior to returning the unit to power following refueling outage 6. The startup was well controlled and accomplished without error. Shift turnovers and briefings were generally thorough and clear (Section O1.1).

- The licensee identified that an operator's failure to ensure the reactor water cleanup (RWCU) leak detection bypass switch was in the bypass position during the performance of loss of offsite power testing caused an inadvertent isolation of the RWCU system (Section O1.2).

- The licensee identified that operating crews did not adequately communicate and control the inoperable condition of a control rod during their shifts and shift turnovers which resulted in a control rod movement prohibited by TS (Section O1.3).

- The licensee identified that a Potential Limiting Condition for Operation was not entered as required due to improper assessment and documentation when the conditions specified in a TS-required step of a surveillance instruction (SVI) were not satisfied. The licensee also identified that the SVI was changed without verifying that the change did not affect past surveillance test results (Section O1.4).

Maintenance

- Overall maintenance activities were effective in improving the material condition of the plant (Section M1.1).

- The safety tag-out for recirculation system flow control valve (FCV) actuator work did not isolate the FCV from the reactor coolant system and a failure of the FCV packing occurred during the actuator work. Several protective barriers in the initiation, authorization, and work release process broke down to produce a potentially hazardous situation for workers. Operators had to respond to minimize a personnel hazard and isolate a reactor coolant leak. Other personnel accumulated radiation dose during the leak recovery actions. This event resulted in a violation of NRC requirements. Another tagging error occurred during restoration of a tag-out which caused an engineered safety feature actuation (Sections M1.2 and M1.3).

- An operator identified that test equipment remained installed on a Reactor Core Isolation Cooling (RCIC) system valve after testing was complete. However, the failure of a maintenance worker to consider the need for environmental qualification

of the valve and to fully communicate the status of the work activity to operations personnel nearly resulted in rendering the RCIC pump inoperable (Section M1.4).

- The licensee identified that an incorrect relay was removed instead of the one specified under a work order. Inadequate self-checking techniques failed to detect a work planning error and caused an initiation of an isolation signal that was an unnecessary challenge to the operators (Section M1.5).

Plant Support

- The fire brigade responded well to smoke in the Service Building elevator (Section F1.1).

Report Details

Summary of Plant Status

The unit remained in its sixth refueling outage until October 20, 1997, when the licensee began a unit startup. The startup was completed on October 23, and power was increased until October 28, when full power was attained. On October 29, power was reduced to about 70 percent to adjust control rod positions and the plant was returned to full power on October 30. The unit operated at full power for the rest of the inspection period except for one minor power reduction for valve testing.

I. Operations

O1 Conduct of Operations

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations.

O1.1 General Comments

a. Inspection Scope (71707)

The inspectors observed many pre-job briefings, shift turnover briefings, and many of the activities that had been discussed at the pre-job briefings. The inspectors also observed preparations for startup from refueling outage 6 (RFO6), and the subsequent startup. Continuous inspection was conducted during the plant startup and initial power increase.

b. Observations and Findings

Shift supervisors and unit supervisors consistently initiated briefings prior to significant plant evolutions. Written briefing summaries were used for almost all briefings. Operations supervisors presented pertinent information to applicable plant personnel during these briefings. The briefings usually involved considerable discussion between team members on responsibilities and expectations. A detailed written plan was developed for the startup, with specific tasks assigned to individuals in advance to allow them to familiarize themselves with task requirements, and in some cases, to practice the task on the simulator. In one case, the specific task description was not adequate (see Section M1.4). Operations personnel were well prepared for startup activities, and kept supervision informed of abnormal conditions.

Three-legged communications were normally followed during RFO6, plant startup, and normal plant operations. The control room appeared crowded at various times during the plant startup. Although no detrimental effects were noted, operations personnel stated that they were periodically challenged by the amount of activity in the control room. There were no operator errors during the plant startup and power ascension.

c. Conclusions

Thorough preparations were made prior to returning the unit to power following RFO6. The startup was well controlled and accomplished without error. Briefings were generally thorough and clear.

O1.2 Reactor Water Cleanup Isolation

a. Inspection Scope (71707)

The inspectors reviewed the circumstances and interviewed personnel involved in the inadvertent isolation of the Reactor Water Cleanup (RWCU) system during the performance of a Surveillance Instruction (SVI).

b. Observations and Findings

During the performance of the Division 1 Loss of Offsite Power (LOOP) Test, SVI-R43-T1337, Revision 1 (March 1994), on October 10, 1997, the RWCU system automatically isolated due to an incorrect bypass switch position. The SVI required the verification of the RWCU leak detection bypass switch in the bypass position. Contrary to this requirement, the operator conducting the verification failed to identify that the switch was actually in the normal position, even though its position was readily visible. Subsequent steps of the SVI initiated an RWCU isolation signal and actuation due to the incorrect position of the bypass switch. There was no actual plant condition that required an RWCU isolation, and the isolation had no potential or actual safety consequences. This personnel error was promptly identified and reviewed by the licensee through its corrective action process. The issue was discussed with operations personnel to curb future self-checking failures. Technical Specification 5.4.1.a specifies, in part, that
written procedures be established, implemented, and maintained covering the applicable procedures recommended in Appendix "A" of Regulatory Guide (RG) 1.33, Revision 2. Technical Specification 5.4.1.a applies to SVI-R43-T1337and the failure to follow the SVI is considered a violation of TS 5.4.1a. This non-repetitive, licensee-identified and corrected violation is being treated as a **Non-Cited Violation (NCV 50-440/97016-01a(DRP))**, consistent with Section VII.B.1 of the NRC Enforcement Policy. The licensee reported this event to the NRC as an engineered safety features (ESF) actuation via the NRC Emergency Notification System (ENS). The licensee appropriately withdrew the report because 10 CFR 50.72 did not require reporting an invalid actuation of an RWCU isolation.

c. Conclusions

An operator's failure to ensure the RWCU leak detection bypass switch was in the bypass position during the performance of LOOP testing caused an inadvertent isolation of the RWCU system.

O1.3 Movement of an Inoperable Control Rod

a. Inspection Scope (71707)

The inspectors reviewed the circumstances associated with and interviewed personnel involved in a control rod movement that had been conducted in violation of the requirements of TS 3.10.4 during control rod drive (CRD) testing.

b. Observations and Findings

On October 14, 1997, operators, with the plant in cold shutdown, were preparing for startup after RFO6. As part of these activities, CRD hydraulic control units (HCUs) had been serviced. Following HCU restoration, control room personnel commenced CRD testing. At approximately 1:00 p.m., control room personnel received an annunciator and indication that the HCU for CRD 18-39 had a scram accumulator leak. The accumulator leak detection equipment was removed from service to permit nitrogen recharging of the accumulator. This rendered the CRD for Rod 18-39 inoperable. During recharging, a leaking instrument fitting was discovered and Instrumentation and Controls (I&C) personnel were called to assist. Day shift operations personnel failed to provide administrative controls for CRD 18-39 by ensuring the condition was deficiency tagged, documented in logs, and turned over to the oncoming crew.

After the 7:00 p.m. shift turnover, rod testing recommenced. At approximately 7:30 p.m., a reactor operator (RO), under direct senior reactor operator (SRO) supervision, withdrew Rod 18-39 from the reactor core approximately 12 inches, then inserted it. All other rods remained fully inserted at that time. The RO and the SRO involved in the rod movement each failed to identify that the CRD was inoperable due to the accumulator leak detection equipment having been removed from service. At approximately 10:00 p.m., I&C technicians informed the RO that the scram accumulator leak detection instrumentation for the CRD 18-39 HCU was isolated. The operators restored the instrumentation to service and noted that accumulator pressure was approximately 1340 pounds per square inch - gauge (psig) with reactor vessel pressure at 0 psig. This was below the TS-required minimum pressure of 1520 psig. The operators then declared the CRD inoperable, initiated a deficiency tag for the leak and initiated a potential issue form (PIF) for the personnel error. The ability to scram the rod is a necessary part of reactivity control that is required to be maintained whenever rods are withdrawn from the core. It was fortuitous that the accumulator pressure was above the actual reactor pressure so that Rod 18-39 could have been scrammed if necessary. The licensee subsequently investigated this event, notified the NRC via the ENS, initiated corrective actions and submitted Licensee Event Report (LER) 97-014. Operations personnel involved were counseled and other operations personnel were briefed to prevent recurrence.

Technical Specification Limiting Condition for Operation 3.10.4 required that scram accumulator pressure be greater than 1520 psig, as referenced in TS 3.9.5, with a rod withdrawn from the core. Contrary to these requirements, control Rod 18-39 was partially withdrawn from the core with scram accumulator pressure less than 1520 psig. This non-repetitive, licensee-identified and corrected violation is being treated as a **Non-Cited Violation (NCV 50-440/97016-02(DRP))**, consistent with Section

VII.B.1 of the NRC Enforcement Policy.

c. Conclusions

The inoperable condition of the HCU for CRD 18-39 was not adequately communicated and controlled by operating crews throughout their shifts and during shift turnover. This allowed a control rod movement which was prohibited by TS. Due to the low pressure in the reactor vessel, the accumulator had sufficient pressure to scram the rod if necessary.

O1.4 Emergency Diesel Generator (EDG) Operability Determination

a. Inspection Scope (37551, 71707 and 92901)

The inspectors reviewed SVI-R43-T5367, "LPCI B and C Initiation and Loss of EH12 Response Time Test," Revision 6 (February 1996) and reviewed the licensee's initial response, investigation, and reporting of a failed TS-required step of the SVI.

b. Observations and Findings

In response to Generic Letter 96-01, "Testing of Safety-Related Logic Circuits," the licensee conducted a review of surveillance instruction SVI-R43-T5367 which tested, in part, the EDG loading sequence during a LOOP or loss of coolant accident (LOCA) event. A procedure reviewer determined that Step 5.1.4.2.i of the SVI, which required verification that Emergency Service Water (ESW) Pump "B" would start within 18 to
22 seconds after EDG breaker closure, should have been marked with a "$" sign signifying that it was a TS-required step. On July 21, 1997, the SVI was changed to include this designation.

On October 12, 1997, during RFO6, SVI-R43-T5367 was performed. When Step 5.1.4.2.i was conducted, ESW Pump "B" started 24.6 seconds after EDG breaker closure, which was outside of the required time period specified in the step. Operations personnel were informed of the failure to satisfy the conditions specified in Step 5.1.4.2.i of the SVI at about 3:30 p.m. Two PIFs were initiated by operations personnel. The first was initiated at about 4:00 a.m. on October 13; however, a TS Limiting Condition for Operation (LCO) or a potential LCO (PLCO) was not initiated nor did operations personnel request an EDG operability determination at that time. A second PIF, 97-2128, was signed by the Shift Supervisor at 9:08 p.m. This PIF initiated PLCO P97-1057, but this PLCO identified the ESW pump as the concern, not the EDG. It was not until the afternoon of October 14, 1997, that a PLCO was initiated for the EDG. Perry Administrative Procedure (PAP) 1105, "Surveillance Test Control," Revision 8 (July 1995), required, in part, that operations personnel take immediate actions to evaluate the operability of equipment and enter the applicable TS LCO when the conditions specified in a "$" denoted step within a TS SVI are not satisfied.

The licensee also determined that the conditions specified in this step had not been

satisfied when the surveillance test was performed during refueling outage 5 (RFO5); however, this was not considered or evaluated by the procedure reviewer when the SVI was revised in July 1997. Perry Administrative Procedure 0522, "Changes to Procedures and Instructions," Revision 8 (June 1996), stated, in part, that the change process would ensure all proposed changes met the criteria described in the In-Depth Review Checklist (IDRC) per PAP-0507, "Preparation, Review, and Approval of Instructions," Revision 11 (June 1996). The IDRC of PAP-0507 required, in part, that the document containing the proposed changes be adequately detailed for verification and sign off of acceptance

criteria, TS acceptance criteria be clearly stated, and that required follow-up actions be taken when the proposed document identified an adverse impact on completed activities.

Technical Specification 5.4.1.a specifies, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Appendix "A" of Regulatory Guide (RG) 1.33, Revision 2. Technical Specification 5.4.1.a applies to PAP-0522, 0507, and 1105. The failure of operations personnel to immediately initiate a TS LCO or PLCO for the Division 2 EDG once the conditions specified in a "$" step of an SVI were not satisfied, is an example of a violation of TS 5.4.1.a in that PAP-1105 required an immediate operability evaluation and entrance into the applicable LCO or PLCO when Step 5.1.4.2.i conditions were not satisfied. The failure of the procedure reviewer to ensure that required follow-up actions were taken when the proposed change to SVI-R43-T5367 resulted in an adverse impact on completed activities (i.e., conditions specified in a "$" step were not satisfied when the subject SVI was performed during RFO5) is an additional example of a violation of TS 5.4.1.a. Corrective actions included both engineering and operations personnel required training on the significance of SVI failures, the need for retrospective reviews when changing procedures, and
the need for prompt equipment operability determinations. This non-repetitive, licensee-identified and corrected violation is being treated as a **Non-Cited Violation (50-440/97016-01b and c (DRP)),** consistent with Section VII.B.1 of the NRC Enforcement Policy.

The Division 1 and 2 EDGs were rated at 7000 kilowatts (kw), which was significantly greater than the load demand on the divisional busses. The EDG maximum expected load during LOOP or LOCA conditions was 5600 kw. Previous engineering evaluations showed that all buss loads could start at time zero without adversely affecting the EDG. Further, ESW pump "B" was not needed for cooling until at least 90 seconds after the EDG started. Therefore, the EDG would not have been adversely effected by the ESW pump loading at 24.6 seconds. The licensee indicated that a review of the SVI to determine if Step 5.1.4.2.i requires a "$" designation would be completed.

c. Conclusions

The lack of a thorough review of SVI-R43-T5367 before its revision resulted in the failure to identify that the conditions specified in a step of the SVI designated as TS-

required, were not satisfied when the SVI was performed during RFO5. In addition, when the conditions specified in this step were not satisfied when the SVI was performed during RFO6, operators failed to recognize that an immediate operability determination and entrance into the applicable LCO or PLCO was needed. These problems resulted in the identification of two examples of a Non-Cited Violation.

O1.5 Operations Staff Resources

Several operations personnel, including two shift supervisors and a shift technical advisor, left the licensee's employment during the inspection period. The operation's department staffing remained above and beyond minimum staffing levels required by NRC regulations and no immediate concerns were noted with the licensee's ability to effectively operate the plant. The licensee evaluated this situation and took several administrative steps to address this situation.

O2 Operational Status of Facilities and Equipment

O2.1 Drywell Closeout

a. Inspection Scope (71707 and 92901)

The inspectors accompanied a plant operations representative for the closeout inspection of the drywell area of the plant.

b. Observations and Findings

The drywell was inspected on October 19, 1997, with no major deficiencies noted. The drywell and suppression pool were well prepared for startup. Some minor debris, such as pieces of duct tape, were discovered by the inspectors during the walkdown and removed by an operations representative. Other minor debris was noted in the suppression pool, and a piece of tape was identified in a safety relief valve cover. These items were removed by equipment cleaners prior to startup.

c. Conclusions

The operational status of facilities and equipment was appropriately addressed by operations personnel prior to drywell closeout.

O7 Quality Assurance in Operations

O7.1 Corrective Action

a. Inspection Scope (71707)

The inspectors evaluated a licensee management initiative to focus attention on timeliness of corrective actions.

b. Observations and Findings

Licensee senior management instructed the corrective action program administrator to maintain a list of the 20 oldest potential issue forms (PIFs) with incomplete corrective actions. The list was included in the handout for the daily managers' meeting and the 10 oldest PIFs were discussed at each meeting. Individuals responsible for completing corrective actions presented their plans for completing the actions and identified areas where they needed assistance. When the initiative began, the oldest PIF was from 1994, at the end of the inspection period the oldest PIF was from 1995.

c. Conclusions

Licensee management's action and oversight were effective in focusing attention on the completion of older corrective actions.

O8 Miscellaneous Operations Issues

O8.1 (Closed) LER 50-440/97-12-00: "*Insufficient Procedural Guidance Results in Reactor Protection System Actuation.*" On September 23, 1997, at about 12:16 a.m., control room operators repositioned the reactor mode switch without realizing that it would cause a reactor protection system actuation. This event was discussed in Inspection Report (IR) No. 50-440/97012. The corrective actions discussed in the LER, including procedure improvements and operator training, are adequate to prevent recurrence.

O8.4 (Closed) LER 50-440/97-14-00: "*Withdrawal of Inoperable Control Rod Results in Operation Prohibited by Technical Specifications.*" This event is discussed in Section O1.3 of this IR.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (61726, 62707, 71500, and 92902)

The inspectors used Inspection Procedures 61726 and 62707 to evaluate several work activities and surveillance tests. The inspectors observed emergent work as well as planned maintenance conducted during the refueling outage, plant startup, and normal operations.

b. Observations and Findings

The activities observed were generally accomplished effectively with appropriate use of drawings and written instructions. Licensee personnel continued to maintain a low threshold in using the PIF process and equipment deficiency tags to identify issues and potential problems. This included examples of personnel identifying their own errors and situations that could contribute to errors or problems that had not yet occurred. The inspectors observed that design changes were implemented to

improve the reliability of the reactor feedwater booster pumps (RFBPs). The pre-job briefing for testing the "C" RFBP and placing it in service was thorough and included clear and detailed communications among operators, maintenance personnel and engineers. Supervisors emphasized the importance of prompt communications of detailed observations, conservative decision making, self checking, and proper preparation. The overall maintenance backlog was reduced and maintained below the licensee's long term goal. An aggressive approach to steam and water leakage improved radiological conditions and reduced the amount of radioactive effluents discharged. The inspectors noted improvements in the maintenance of work records during the conduct of work, and as a result less effort was required near the end of RFO6 to gather missing information to close out work documents.

c. Conclusions

Although exceptions are discussed in this report, overall maintenance activities were effective in improving the material condition of the plant.

M1.2 Poor Safety Tagging Led to a Reactor Coolant Leak

a. Inspection Scope (62707, 71707, and 92902)

The inspectors reviewed the circumstances surrounding the reactor recirculation system flow control valve (FCV) packing failure during FCV actuator work.

b. Observations and Findings

On October 6, 1997, with the plant in cold shutdown, contract maintenance workers were sprayed with reactor coolant as they worked on the actuator for the "A" Reactor Recirculation FCV in the drywell. The workers were contaminated but no appreciable dose was received and no personnel injuries occurred as a result of this event. Safety tag-out 27868 for work on the "A" FCV did not isolate the work area from the reactor coolant system (RCS). Before beginning work, the workers asked their supervisor if the work area needed to be isolated from the reactor coolant system. The supervisor assumed that the relevant piping was still drained as it had been the previous day and indicated to the workers that the tag-out was proper. However, the piping had been refilled and was open to the reactor coolant system. During the actuator work, the FCV packing cartridge failed and the workers were sprayed with water. A non-licensed operator observed the water spray (estimated at 100 gallons per minute (GPM)) and notified the control room. The control room operators promptly closed the maintenance valves for the recirculation loop and the leakage was reduced to about 10 GPM. The operators did not observe any RCS level decrease.

Technical Specification 5.4.1.a specifies, in part, that written procedures be implemented covering the applicable procedures recommended in Appendix "A" of RG 1.33,
Revision 2. Appendix "A" of RG 1.33 recommended that safety tagging be implemented by a written procedure. Perry Administrative Procedure-1401, "Safety Tagging,"

Revision 8 (January 1995), required, in part, that tag-outs be prepared and verified to adequately isolate potential hazards to personnel and equipment prior to the commencement of work.

Safety tag-out 27868 was not adequately prepared and verified to isolate personnel or equipment from the potential hazards associated with the flow control valve actuator work. This was a violation **(VIO 50-440/97016-03(DRP))** of TS 5.4.1.a. Although this was a licensee-identified and corrected violation, it did not meet the requirements for enforcement discretion of Section VII.B.1 of the NRC Enforcement Policy because it was a repetitive violation. Violation 50-440/97007-01b(DRP), identified on June 2, 1997, occurred because licensee personnel did not adequately verify that a tag-out adequately isolated a potential hazard to personnel. Also, in the recent past there have been several plant events and problems that have occurred because of poor communications. The investigation for PIF 97-1962, which was initiated for this event, identified 12 contributing factors for this violation; 7 involved poor communications. In addition to the personnel and equipment hazard associated with this event, additional personnel radiation dose was accumulated during the cleanup of the drywell that was required as a result of the spilled reactor coolant.

c. Conclusions

The safety tag-out for recirculation system FCV actuator work did not isolate the FCV from the reactor coolant system and a failure of the FCV packing occurred during the actuator work. Several protective barriers in the initiation, authorization, and work release process broke down to produce a potentially hazardous situation for workers. Operators had to respond to minimize a personnel hazard and isolate a reactor coolant leak. Other personnel accumulated radiation dose during the leak recovery actions.

M1.3 Poor Control of Safety Tagging Caused ESF Actuation

a. Inspection Scope (62707 and 92902)

The inspectors reviewed the licensee's evaluation of an ESF actuation that was caused by incorrect sequencing of a restoration from a safety tag-out.

b. Observations and Findings

On October 9, 1997, with the plant shut down during refueling activities, operators were removing safety tags from the CRD hydraulic system. Perry Administrative Procedure-1401, "Safety Tagging," Revision 8 (January 1995), Step 6.4.13, requires that the tag-out reviewer consider the need to specify an order to be followed when removing tags. The SRO in charge of removing the tags and restoring the CRD hydraulic system (reviewer) did not adequately consider the order of removing the tags for tag-out 27835 and when the valves for the CRD hydraulic system were restored to their normal position, normal leakage filled the scram discharge volumes until a high scram discharge volume scram occurred. All rods were already fully

inserted so there was no rod motion. The personnel involved were counseled. The failure to properly use written instructions appropriate to the circumstances for this work was an additional example of a TS 5.4.1.a violation. The corrective actions for previous tagging errors could not have reasonably been expected to have prevented this event from occurring. Therefore, this non-repetitive, licensee-identified (as a result of a self-revealing event), and corrected violation is being treated as a **Non-Cited Violation (50-440/97016-01d(DRP))**, consistent with Section VII.B.1 of the NRC Enforcement Policy.

c. Conclusions

Personnel errors in restoring a safety tag-out caused an ESF actuation and resulted in the identification of an additional example of a TS 5.4.1.a Non-Cited Violation.

M1.4 Improper Control of Test Equipment

a. Inspection Scope (61726, 62707, and 92902)

The inspectors reviewed the actions associated with the failure to remove test equipment from a reactor core isolation cooling (RCIC) system motor operated valve (MOV) following a test.

b. Observations and Findings

Post-outage RCIC system testing included motor operated valve (MOV) testing during both cold and hot conditions. On October 20, 1997, the limit switch cover for MOV 1E51-F0019 was removed to allow the installation of test equipment for the cold test. Once the cold test was completed, rather than remove the equipment, the technician left it in place due to the need to perform an additional test on the MOV at hot conditions. However, in the time between the two tests, the reactor pressure was to be raised above 200 psig, the pressure above which environmental qualification (EQ) of the valve is needed. The MOV limit switch cover was required to be installed to assure EQ of the valve. The decision to leave the cover off, and the potential inoperability of the valve, were not adequately communicated to operations. Prior to the reactor pressure reaching 200 psig, this condition was identified by an operator performing rounds in the area. The test equipment was removed and the MOV limit switch cover was installed. A subsequent safety evaluation determined that the RCIC pump was operable because EQ for the valve was not needed for the plant conditions at the time this condition was identified. It was fortuitous that the timing of the operator identifying this problem coincided with reactor pressure being below 200 psig.

c. Conclusions

The failure of a maintenance worker to consider the need for environmental qualification of MOV 1E51-F0019 and to fully communicate the status of the work activity to operations personnel nearly resulted in rendering the RCIC pump inoperable. However, due to the discovery and removal of the test equipment prior

to the reactor reaching 200 psig, the RCIC pump remained operable.

M1.5 Incorrect Relay Replaced

a. Inspection Scope (61726, 62707, and 92902)

The inspectors reviewed the actions associated with the replacement of an incorrect relay under Work Order (WO) 97-1918.

b. Observations and Findings

On November 20, 1997, relay 1B21H-K4C (labeled "CK") was removed instead of relay 1C71A-K4C (labeled "CD"). Due to a planning personnel error, the WO incorrectly identified the C71A-K4C relay as "CK." The removal of the incorrect relay resulted in a half logic actuation of the main steam line isolation function.

Because of inadequate self checking techniques, maintenance personnel failed to detect the work planning error. The event was promptly identified, the correct relay was replaced, and the isolation was reset. This non-repetitive, licensee-identified and corrected violation is an additional example of a TS 5.4.1.i violation and is being treated as a **Non-Cited Violation (50-440/97016-01e(DRP))**, consistent with Section VII.B.1 of the NRC Enforcement Policy.

c. Conclusions

Inadequate self-checking techniques failed to detect a work planning error and caused an initiation of an isolation signal that was an unnecessary challenge to the operators.

M8 Miscellaneous Maintenance Issues (92700)

M8.1 (Closed) LER 50-440/97-007-00: *"Loss of Electrical Power to Reactor Protection System Bus Due to Electrical Protective Assembly Trip Results in Engineered Safety Feature Actuations."* On July 13, 1997, at about 11:58 a.m., electrical power from the Division 2 normal power source to Reactor Protection System Bus "B" was lost. This event was discussed in Inspection Report (IR) No. 50-440/97009. The cause of the event was determined to be unreliable operation of the electrical protective assembly logic control board. This problem was similar to that reported in LER 97-003-00. Completion of corrective actions will be evaluated during the inspectors' review of LER 97-003-00.

M8.2 (Closed) LER 50-440/97-01[illegible] *"Loss of Electrical Power to Reactor Protection System Bus Due to Electri[illegible] [illegible]tective Assembly Trip Results in Engineered Safety Feature Actuations."* This re[illegible]ion to LER 97-010-00 corrected an error the licensee identified in the original LER which incorrectly stated that the event caused a RCIC isolation. There was no RCIC isolation. Therefore, the event had slightly less potential safety consequences than originally indicated. Licensee Event Report 97-010-00 was closed in IR No. 50-440/97012 because the corrective actions for LER

97-010-00 will be evaluated during the inspector's review of LER 97-003-00. This revision had no impact on that planned review.

M8.3 (Closed) LER 50-440/97-011-00: *"Technical Specification Surveillance Test Performance Results in Engineered Safety Feature Actuations."* On September 21, 1997, at about 4:53 a.m., with the plant shutdown during refueling operations, I&C technicians performing a surveillance test caused an inadvertent pressure transient in the reference leg for two level instruments. The pressure transient caused a false reactor pressure vessel low water level ESF actuation. All safety equipment operated as required for the existing plant conditions and there was no adverse effect on plant equipment. The corrective actions discussed in the LER, including procedure improvements and I&C technician training, are adequate to prevent recurrence.

M8.4 (Open) LER 50-440/97-013-00: *"Control Rod Drive Hydraulic System Maintenance Activities Result in Reactor Protection System Actuations."* This LER reported two similar events regarding safety tagging of the control rod drive hydraulic system. The event that involved an ESF actuation on October 9, 1997, is discussed in Section M1.3 of this inspection report. No additional inspection is required for that event. The event that occurred on October 11, 1997, is the subject of IR No. 50-440/97022.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Suppression Pool Level Indication

a. Inspection Scope (37551, 61726, and 92903)

The inspectors reviewed the licensee's initial evaluation of larger than expected suppression pool level indication oscillations during a high pressure core spray (HPCS) system surveillance test.

b. Observations and Findings

During a surveillance test of the HPCS system, which directed system flow to the suppression pool via the test return line, control room operators observed that suppression pool level indication oscillations were larger than had been observed in the past. Operators dispatched to the containment to observe the surface of the suppression pool determined that the oscillations were not as large as the instrumentation indicated. The operators documented their observation with PIF 97-2168. During a discussion with the inspectors, engineers stated that the newly installed emergency core cooling systems strainer in the suppression pool appeared to be causing larger pressure oscillations which had been indicated as level oscillations by the pressure differential level indication.

In performing its safety function, HPCS would not be returning flow to the

suppression pool, as it did in the test, so the strainer would not impact the HPCS safety function. The inspectors and the licensee monitored suppression pool level indications during RCIC operation and safety relief valve (SRV) testing because RCIC and the SRVs discharge steam to the suppression pool when they are required to perform their safety functions. The inspectors noted only minor variations in level indication during RCIC and SRV operations. The licensee performed another HPCS test with temporary video cameras in the containment and verified that actual suppression pool levels were not fluctuating more than expected. The licensee completed its investigation of PIF 97-2168 and concluded that the design of the level instrument sensing lines allowed an undesirable accumulation of air in the lines. The licensee concluded that the lines could be vented sufficiently to maintain the operability of the level instruments. However, the licensee developed eight corrective actions for the PIF, which included development of a more thorough venting method and evaluation of a modification to the sensing lines. The inspectors will review the implementation of the corrective actions for this phenomenon during a future inspection (**IFI 50-440/97016-04(DRP)**).

c. Conclusions

The accumulation of air in the suppression pool level instrument sensing lines led to indications that level oscillations during a HPCS system surveillance test were larger than they actually were.

E2.2 RCIC Governor Valve

a. Inspection Scope (37551 and 92903)

The inspectors reviewed the trip and troubleshooting of the RCIC turbine during startup after RFO6.

b. Observations and Findings

During plant startup and heat up, the RCIC turbine was operated for Inservice Inspection (ISI). The cold start of the turbine for the ISI was normal, as were other cold starts. Near the end of the ISI, the RCIC turbine unexpectedly tripped, possibly as a result of its controls being gently bumped by personnel working on the turbine. During two immediate attempts to restart the hot turbine, it also tripped. The licensee delayed plant startup to disassemble the governor valve and thoroughly investigate the cause of the trips. The investigation revealed that the manufacturing tolerances of the turbine governor valve stem and carbon spacer rings did not match design tolerances. The stem was replaced with a component that was acceptable, but was more susceptible to a previously exhibited corrosion problem. The licensee completed its corrective action program investigation near the end of the inspection period and identified several corrective actions. The inspectors need to evaluate the past operability of the RCIC turbine, the adequacy of corrective actions, the adequacy of the surveillance testing methods, and the root cause of the failed stem and spacer rings. This will remain an **Unresolved Item (URI 50-440/97016-05(DRP))** until the inspectors complete their evaluation.

c. Conclusions

Following unexpected trips of the RCIC turbine, the licensee promptly completed a thorough investigation to ensure the RCIC system was operable. Issues identified during the investigation require further NRC review.

IV. Plant Support

F1 Fire Protection Staff Knowledge and Performance

During plant startup, the fire brigade responded to the report of smoke from the service building elevator. The brigade responded within 2 to 3 minutes and immediately discovered the source of smoke was the elevator motor. Appropriate equipment was deenergized, and steps were carried out promptly to secure the area. No fire was observed. The fire brigade was knowledgeable of the proper actions to take and demonstrated that their response training was effective.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on December 1, 1997. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

L. W. Myers, Vice President, Nuclear
W. R. Kanda, General Manager Nuclear Power Plant Department
T. S. Rausch, Director, Quality and Personnel Development Department
N. L. Bonner, Director, Nuclear Maintenance Department
R. W. Schrauder, Director, Nuclear Engineering Department
H. W. Bergendahl, Director, Nuclear Services Department
J. Messina, Operations Manager
J. T. Sears, Radiation Protection Manager
F. A. Kearney, Superintendent Plant Operations

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71500:	BOP
IP 71707:	Plant Operations
IP 71714:	Cold Weather Preparations
IP 71750:	Plant Support Activities
IP 92700:	Onsite Follow-up of Written Reports of Non-routine Events at Power Reactor Facilities
IP 92901:	Follow-up - Plant Operations
IP 92902:	Follow-up - Maintenance
IP 92903:	Follow-up - Engineering
IP 92904	Follow-up - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-440/97016-01a(DRP)	NCV	Inadvertent RWCU Isolation
50-440/97016-01b(DRP)	NCV	Entering Incorrect PLCO
50-440/97016-01c(DRP)	NCV	Improper Procedure Change
50-440/97016-01d(DRP)	NCV	Improper Safety Tag Restoration
50-440/97016-01e(DRP)	NCV	Incorrect Relay Replacement
50-440/97016-02(DRP)	NCV	Movement of Inoperable Control Rod
50-440/97016-03(DRP)	VIO	Improper Safety Tagging
50-440/97016-04(DRP)	IFI	Suppression Pool Level Concerns
50-440/97016-05(DRP)	URI	Unexpected RCIC Turbine Trips

Closed

50-440/97016-01a(DRP)	NCV	Inadvertent RWCU Isolation
50-440/97016-01b(DRP)	NCV	Entering Incorrect PLCO
50-440/97016-01c(DRP)	NCV	Improper Procedure Change
50-440/97016-01d(DRP)	NCV	Improper Safety Tag Restoration

50-440/97016-01e(DRP)	NCV	Incorrect Relay Replacement
50-440/97016-02(DRP)	NCV	Movement of Inoperable Control Rod
50-440/97-007-00	LER	Loss of Electrical Power to Reactor Protection system Bus Due to Electrical Protective Assembly Trip Results in engineered Safety Feature Actuations
50-440/97-010-01	LER	Loss of Electrical Power to Reactor Protection system Bus Due to Electrical Protective Assembly Trip Results in Engineered Safety Feature Actuations
50-440/97-011-00	LER	Technical Specification Surveillance Test Performance Results in Engineered Safety Feature Actuations
50-440/97-012-00	LER	Insufficient Procedural Guidance Results in Reactor Protection System Actuation
50-440/97-014-00	LER	Withdrawal of Inoperable Control Rod Results in Operation Prohibited by Technical Specifications

Discussed

50-440/97-013-00	LER	Control Rod Drive Hydraulic System Maintenance Activities Result in Reactor Protection System Actuations

LIST OF ACRONYMS AND INITIALISMS

CFR	Code of Federal Regulations
CRD	Control Rod Drive
EDG	Emergency Diesel Generator
ENS	Emergency Notification System
EPA	Electrical Protective Assembly
EQ	Environmental Qualification
ESF	Engineered Safety Feature
ESW	Emergency Service Water
HCU	Hydraulic Control Unit
HPCS	High Pressure Core Spray
I&C	Instrumentation and Control
IDRC	In-Depth Reviewer Checklist
IFI	Inspector Followup Item
IR	Inspection Report
KW	Kilowatts
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PAP	Perry Administrative Procedure
PDR	Public Document Room
PIF	Potential Issue Form
PLCO	Potential Limiting Condition for Operation
PSIG	Pounds per Square Inch, Gage
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RFBP	Reactor Feedwater Booster Pump
RFO5	Refueling Outage 5
RFO6	Refueling Outage 6
RG	Regulatory Guide
RI	Resident Inspector
RO	Reactor Operator
RWCU	Reactor Water Cleanup
RPS	Reactor Protection System
SRI	Senior Resident Inspector
SRO	Senior Reactor Operator
SRV	Safety Relief Valve
SVI	Surveillance Instruction
TS	Technical Specification
URI	Unresolved Item
USAR	Updated Safety Analysis Report
VIO	Violation
WO	Work Order