U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	50-440/97004
Licensee:	Centerior Service Company
Facility:	Perry Nuclear Power Plant
Location:	P. O. Box 97, A200 Perry, OH 44081
Dates:	March 22 - May 2, 1997
Inspectors:	D. Kosloff, Senior Resident Inspector R. Twigg, Resident Inspector
Approved by:	C. G. Miller, Chief, Projects Branch 4 Division of Reactor Projects

EXECUTIVE SUMMARY

Perry Nuclear Power Plant, Unit 1
NRC Inspection Report 50-440/97004

This inspection included aspects of licensee operations, maintenance and surveillance, engineering, and plant support. The report covers a 6-week period of resident inspection.

Operations

Recent and longer term improvements in the control room working environment were maintained, and additional improvements were also effective (Section O1.2).

Operator response to a partial reactor trip signal and partial containment isolation were appropriate, as were the planned corrective actions (Section O2.1).

Operations appropriately controlled a transient that slowly reduced plant power. An incorrect system operating instruction (SOI) that led to the transient was an example of poor procedural support for operations. There was a missed opportunity to catch the SOI error prior to the transient. A flow control valve surveillance test was adequate, although improvements were possible (Section O3.1).

Operators performed well during two feedwater system evolutions. Supervisors used appropriate command and control methods for both evolutions. The briefing for one evolution was very good and the briefing for the other was excellent (Section O3.2).

Improvements continued in shift briefings at turnover and in plant evolution briefings. The inspectors observed several excellent briefings (Section O4.1).

A shift supervisor reviewed an operability evaluation with a critical questioning attitude and the operations manager appropriately performed his oversight function (Section O4.2).

Maintenance and Surveillance

Maintenance and surveillance activities were appropriately performed. Operators exhibited good questioning attitudes and engineering support was appropriate (Section M1.1).

Overall, past improvements in plant housekeeping and material condition had been maintained. However, two operator workarounds had caused increased operator burden (Section M2).

Engineering

Review of an emergency diesel generator surveillance indicated that there was a weakness in understanding of the design basis. A more detailed review indicated that the surveillance was acceptable (Section E4.1).

Plant Support

A fire protection technician displayed awareness of the safety significance of the test he was performing and thorough knowledge of the fire protection system's capabilities to protect the reactor (Section F5).

Report Details

Summary of Plant Status

The plant operated at or near full power throughout most of the inspection period. On April 6, 1997, power was reduced to about 73% to adjust control rod positions. On April 9, 1997, power dropped to about 97% power during a slow transient as reactor recirculation flow control valves drifted in the closed direction. In both cases the plant was restored to full power the same day.

I. Operations

O1 Conduct of Operations[1]

O1.1 General Comments (71707)

During the inspection period one event occurred which required prompt notification of the NRC pursuant to 10 CFR 50.72. On March 22, 1997, a reactor protection system electrical protective assembly trip caused a partial reactor trip signal and a partial containment isolation.

O1.2 Control Room Inspections and Plant Area Walkdowns (71707)

a. Inspection Scope (71707, 92901)

The inspectors performed frequent routine inspections in the control room and throughout the plant.

b. Observations and Findings

During the inspection period, the licensee moved work control activities previously conducted in the Unit 1 control room to the adjacent Unit 2 (abandoned) control room. The shift technical advisors, who hold senior reactor operator licenses, were given responsibility for the detailed administrative review of plant work. An additional experienced unit supervisor was temporarily assigned to assist the STAs in taking on this new responsibility. This action reduced personnel traffic in the control room and allowed the shift supervisors and unit supervisors to increase their direct oversight and supervision of plant operations. The unit supervisors continued direct supervision of all planned or potential reactivity changes.

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[1] Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

The inspectors noted that all observed communications involving operators continued in strict compliance with the licensee's three-part communications rule (initial communication, repeat back, acknowledgement). This included communications with personnel in other organizations and communications outside the control room that were monitored on the control room radio. The inspectors also noted that communications not involving operators also often involved the three-part method. The inspectors observed that all control room annunciators continued to be called out and acknowledged by the operators. The inspectors also observed that planned procedures were reviewed by operators prior to use and were readily available and used during conduct of activities or when intermittent or imminent use was expected. Appropriate controls were maintained for personnel access to the at-the-controls area.

While inspecting containment, the inspectors identified that one of the safety related motor operated suppression pool makeup valves (G43-F040B) was allowing water to leak from the containment upper pool to the suppression pool. The inspectors reported the leakage to the unit supervisor who initiated a deficiency tag for G43-F040B. About 1 week later, the inspectors verified that the leakage had been stopped. All other leaks observed in the plant had been identified by the licensee.

c. Conclusions

Recent and longer term improvements in the control room working environment were maintained, and additional improvements were also effective. Although routine operator and supervisory plant tours did not identify a leak from the upper containment pool to the suppression pool, this was an isolated case.

O2 Operational Status of Facilities and Equipment

O2.1 Equipment Failure Caused Partial Reactor Trip Signal and Partial Containment Isolation

a. Inspection Scope (37551, 62707, and 71707)

The inspectors evaluated the operators' response to the failure of a reactor protection system (RPS) electrical protective assembly (EPA). The inspectors reviewed troubleshooting conducted by maintenance and engineering. The inspectors also reviewed General Electric (GE) Service Information Letter (SIL) No. 496 (August 23, 1989), "Electrical Protection Assembly Performance," SIL No. 496, Rev. 1 (September 14, 1990), SIL No. 496, Rev. 1, Supplement 1 (October 12, 1995), and the licensee's evaluations of the SIL.

b. Observations and Findings

On March 22, 1997, power was lost to RPS Bus 'B.' This caused a partial reactor trip signal and a partial containment isolation of reactor water cleanup and main steam line drains. All equipment functioned as expected. The operators transferred

RPS Bus 'B' to its alternate supply, reset the partial reactor trip signal, and restored the isolated equipment.

Engineering and maintenance attempted to determine the cause of the power loss. Although an RPS power supply EPA had tripped, there was no indication that instability of the power supply had caused the trip. The failed EPA was replaced with a spare and sent to GE, the vendor, for analysis. The licensee was still monitoring the RPS 'B' normal power supply at the end of the inspection, and no abnormalities had been observed. GE found a fault in the EPA integrated circuit chips which had probably caused the trip. The inspectors reviewed the SILs and Licensee Event Report (LER) 92-001, which described the previous EPA trip. The inspectors also reviewed LER 97-003, which described the March 22 EPA trip. The LER included a commitment to replace all existing EPA logic control boards with a newer version, model no. 148D611G002, by June 30, 1998. The inspectors informed the licensee that the model no. for the newer version board in the LER was incorrect, there were no EPAs with the model no. stated in the LER. The inspectors verified that the licensee had changed the model no. in its commitment tracking and corrective action systems to the correct model no. 148C6118G002.

c. Conclusions

Operator response to this event and engineering response to the SILs was appropriate, as were the planned corrective actions. The undetected errors in the licensee's LER and commitment documentation indicated a lack of attention to detail.

O3 Operations Procedures and Documentation

O3.1 Flow Control Valve Motion Caused Slow Plant Transient

a. Inspection Scope (37551, 61726, 71707, and 92902)

The inspectors observed operators control the plant during a maintenance activity that required the reactor recirculation (B-33) flow control valve (FCV) hydraulic power units (HPU) to be stopped. The inspectors also observed the plant evolution briefing, the maintenance activity, preparation of a temporary instruction change, and the engineering evaluation of the unexpected transient.

b. Observations and Findings

The unit supervisor conducted a thorough briefing of maintenance and operations personnel who were going to participate in the verification of proper fuses in an average power range monitor (APRM) power supply and related FCV operations. The FCV HPUs were stopped, immobilizing ("locking up") the FCVs. The APRM fuses, supplied by the power supply vendor, were the wrong type, and were promptly replaced. The operators then began to start the HPUs using System Operating Instruction (SOI) B-33, "Reactor Recirculation System." Voltages and currents measured as required by the SOI were not as expected, and the operators determined that a step that had been added a few days before was out of sequence. The operators requested that the SOI be revised. While the procedure change was being completed the FCVs remained locked up. The B-33 responsible system engineers (RSE), who had come to the control room to observe the HPU starts, and the operators noted that the FCVs were slowly drifting closed. In about 3.5 hours one FCV moved about 11% and the other moved about 5%. This caused a slow reduction in reactor power. The operators contacted reactor engineering and were informed that no thermal limits would be approached in maneuvering the plant as long as power did not drop below 90%. The inspectors observed that power dropped to about 97% before the revised SOI could be used to start the HPUs. The unit supervisor conducted another thorough briefing and the HPUs were started with no further problems. The operators noted that Technical Specification (TS) 3.4.2 required the FCVs to fail "as-is" upon loss of hydraulic pressure and initiated Potential Issue Form (PIF) 97-0622 because they did not equate 11% movement (.05% per minute) with failing "as-is." The RSEs had questioned GE about this condition earlier and GE had indicated that the FCVs could move up to 15% per minute without invalidating the loss of coolant accident (LOCA) analysis. The LOCA analysis was the basis for the TS requirement for the FCVs to fail "as-is." The inspectors' review of the USAR and TS basis did not reveal anything that was inconsistent with the GE position.

The inspectors reviewed surveillance instruction (SVI) B33-T1158, which verified that the FCVs fail as-is, and confirmed that the SVI methodology would verify that the valves were not drifting at greater than 15% per minute although the test would not provide a specific drift rate. The inspectors discussed this with the RSEs who stated that the FCV actuators are checked during refueling outages for drift rate and would be repaired or replaced at a drift rate much lower than 15%. The surveillance requirement for "as-is" verification is a refueling interval surveillance. The inspectors pointed out that the SVI could be misinterpreted as allowing the FCVs to move for a few seconds instead of stopping immediately. This could happen if the pilot operated isolation valves failed when HPU power was removed because it would take a few seconds for the HPU accumulators to lose pressure and the FCV would then fail as-is even though the pilot operated isolation valve had failed. The RSEs stated that they would evaluate the SVI to see if changes would be appropriate to avoid the possibility of a misinterpretation.

c. Conclusions

The inspectors concluded that operations appropriately controlled the maintenance activity and plant transient with appropriate support from other organizations. The incorrect change to SOI B-33 challenged the operators and was another example of poor procedural support for operations. Some past examples have been identified by the operators and others have caused plant transients or violations of NRC requirements. Operations and maintenance personnel missed an opportunity, albeit limited, to catch the SOI error during the evolution briefing. The inspectors concluded that an FCV SVI was adequate as written although improvements were possible.

O3.2 Feedwater Evolutions to Compensate for Equipment Failures

a. Inspection Scope (71707)

The inspectors observed the operators prepare for and perform two feedwater (FW) system evolutions. One was to take leaking moisture separator reheaters (MSR) out of service, and the second was to take alternate control of level for a FW heater.

b. Observations and Findings

The MSR and FW drain tank levels were normally controlled automatically with their primary level control valves. However, after a robot was used to locate and evaluate leaks on MSR drain tanks, the operators were required to take the MSRs out of service. Also, the primary level control valve failed on the 3B FW heater and the operators were required to place the alternate level control valve in control. Detailed formal briefings were conducted for both evolutions using standard pre-job briefing checklists prepared for each briefing. The start and stop of the 3B FW heater briefing was well defined. The applicable system operating instructions were reviewed during each briefing. Potential reactivity effects and precautions were discussed. Malfunctions that could occur during the evolutions were discussed with reference to the applicable off normal instructions.

Both evolutions were performed in a deliberate manner with complete and clear communications. During the MSR evolution, the operators became concerned about exceeding the cooldown rate for the MSR and suspended the evolution to develop a temporary Operations Evolution Order (OEO) to improve the control of the cooldown. Operations completed the evolution and then recognized that the OEO had not been needed because the maximum temperature reduction possible for the condition of the equipment would have prevented the cooldown rate from being exceeded. The operators recommended that information be added to the instruction to indicate when the cooldown rate would not be a concern.

c. Conclusions

The operators performed well during both evolutions. The shift and unit supervisors used appropriate command and control methods in preparing for and supervising both evolutions. The briefing for the MSR evolution was very good. The briefing for the 3B FW Heater evolution was excellent.

O4 Operator Knowledge and Performance

O4.1 Shift Turnover and Evolution Briefings

a. Inspection Scope (71707)

The inspectors observed many shift turnover and plant evolution briefings during the inspection period.

b. Observations and Findings

The following positive attributes were often observed during the briefings:

- Discussions of the days work, plant equipment problems, and resources available were extensive, with all members of the crew actively engaged. Supervisors would sometimes ask individuals specific questions to encourage participation.

- Questioning attitudes were demonstrated by various personnel throughout the briefings. Supervisors welcomed questions and often used the questions to emphasize additional information.

- Briefing participants were reminded, when appropriate, to speak to the whole crew and not just the supervisors.

- Training sessions at the beginning or end of the turnover briefings engaged the crew actively with realistic examples and discussion.

- A specific training session for each crew during turnover briefings involved discussion of an event at another plant with written questions regarding Perry procedures that would prevent a similar event at Perry. The questions were answered in writing by each crew member.

- The plant operators were periodically reminded to closely observe contract workers removing Thermolag and to ensure compliance with general plant work guidelines. The operators were also reminded to monitor the use and condition of associated scaffold, with various specific examples provided based on supervisors' observations of work in progress.

- Plant evolution briefings were prepared in advance using a standard briefing format with specific topics listed. This included a reinforcement of the need to identify any procedure problems and safely stop activities if the procedure was incorrect or unclear.

c. Conclusions

In the past, shift briefings at turnover had not stood out either negatively or positively. Also, there had been few plant evolution briefings. The quality of briefings had improved during the last inspection period. In general this improvement continued. Several briefings during this inspection period, that included most of the attributes discussed above, were excellent.

O4.2 Operations Response to a Failed Surveillance

a. Inspection Scope (37551, 61726, and 71707)

The inspectors observed operations management, shift supervisors, and engineering personnel review a failed surveillance (SVI-M14-T9313, "Type C Local Leak Rate Test of 1M14 Penetration P313") for operability. The surveillance tested the containment and drywell purge system supply penetration, used during normal power operations for containment ventilation.

b. Observations and Findings

During performance of the surveillance, the 18-inch inboard M14-F0195 valve (in parallel with a 42-inch valve used during shutdown conditions) failed to meet its acceptance criteria, a leakage limit of 4310 standard cubic centimeters per minute (SCCM). The actual, as-found leakage was 7440 SCCM. With M14-F0190 (18-inch inboard valve in series with M14-F0195) closed, leakage was only 23.66 SCCM. The shift supervisor discussed operability concerns with engineering personnel and concluded that M14-F0190 would be isolated and controlled administratively until M14-F0195 was repaired (priority 3, work within 3 weeks). The operations manager then reviewed the situation with the shift supervisor and engineering personnel. Numerous questions were discussed, with the decision unchanged.

c. Conclusion

The shift supervisor performed well with a critical questioning attitude and the operations manager appropriately performed his oversight function.

O8 Miscellaneous Operations Issues (71707 and 92700)

O8.1 Review of Institute of Nuclear Power Operations Assessment

The inspectors reviewed the 1996 Institute of Nuclear Power Operations (INPO) evaluation of Perry to determine if there were any safety issues which were previously unknown to the NRC. The INPO report documented findings of similar programmatic problems to those previously identified by the NRC and the licensee.

O8.2 (Closed) LER 50-440/97-03-00: "Loss of Electrical Power to Reactor Protection System Bus Due to Electrical Protective Assembly Trip Results in Engineered Safety

Feature Actuation." The licensee's corrective actions are discussed in Section O2.1 and the LER is closed based on the review described in that section.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (60705, 61726, 62707, and 92902)

The inspectors observed all or portions of the following work and surveillance instruction (SVI) activities with no concerns identified. Additional items are discussed under Observations and Findings.

Polarity check of breakers EF1B10 and EF1D10
Refurbishment of Reactor Feed Booster Pump 'C' (work not completed)
Repair of the Unit 1 service air compressor (work not completed)
Removal of Thermolag (work not completed)
Replacement of average power range monitor fuses (See Section 3.1)
Preparation of refueling equipment for refueling

SVI-B33-T1158, "Reactor Recirculation Flow Control Valve Functional Test" (See Section O3.1)
SVI-C11-T1003-B, "Control Rod Exercise (Part 2)"
SVI-E22-T1339, "Division III High Pressure Core Spray Emergency Diesel Generator 18-month Loss Of Off-site Power Test" (See Section E4.1)
SVI-M14-T9313, "Type C Local Leak Rate Test of 1M14 Penetration P313" (See Section O4.2)
SVI-M15-T5417, "Annulus Exhaust Gas Treatment System Automatic Initiation"

b. Observations and Findings

During performance of SVI R43-T1317, "Emergency Diesel Generator Monthly Test - Division I," the non-licensed operators observed that small amounts of fluid were ejected from two cylinder heads during pretest air roll. Engineering and operations personnel responded promptly to the EDG room, and concluded that the amount of oil was not abnormal. No water was identified. The test was completed with no problems.

The inspectors performed a detailed review of surveillance instruction, SVI-E31-T0075-B, "Main Steam Line High Flow Channel B Calibration for E31-N088B." This included a review of the TS basis, the USAR, and associated drawings. The instruction was clearly written with appropriate controls to prevent an inadvertent scram. The return to service section of the SVI required independent verification that the flow transmitter had been correctly restored.

c. Conclusions

Maintenance and surveillance activities were appropriately performed. Operators exhibited good questioning attitudes, and engineering support was appropriate.

M2 Maintenance and Material Condition of Facilities and Equipment

a. Inspection Scope (71707, 92720)

The inspectors observed plant conditions during plant walk downs.

b. Observations and Findings

During inspector walkdowns, observations indicated that past improvements in plant housekeeping and material condition had been maintained.

As discussed in the previous inspection report, operators continued to periodically vent the residual heat removal (RHR) suction and discharge headers due to leaking isolation valves. Although the leakage was within design limits, the venting was considered an operator work around. Engineering was evaluating alternate venting and fill methods that could reduce the operator burden and radiation dose. Plant management was also evaluating the need for a plant outage to repair the leaking valves.

c. Conclusions

Overall, past improvements in plant housekeeping and material condition had been maintained. However, the operator workarounds related to RHR valve leakage had caused increased operator burden.

M8 Miscellaneous Maintenance Issues (62707, 61726, 92700, and 92902)

M8.1 (Closed) LER 50-440/94-001-00: "Reactor Water Cleanup (RWCU) Isolation Due to Loss of Auxiliary Building Ventilation." On January 18, 1994, system containment isolation signals occurred due to RWCU system valve nest room high differential temperatures. Prior to the signals, the operators responded to the high differential temperature alarms by isolating the RWCU system. The high differential temperatures occurred because the auxiliary building ventilation system supply fans tripped when sensors indicated the inlet air temperature was too low. One of three sensors had failed and, the capillary tube for the three sensors was not located properly to give a true indication of inlet air temperature. The inspectors verified that the failed sensor was replaced and the capillary tube was repositioned.

M8.2 (Closed) LER 50-440/94-007-00: "Water Intrusion Leads to Passive Seismic Instrument Failure." During February 17, 1994, performance of a surveillance on the sealed passive seismic monitoring system instrument located on the floor of the high pressure core spray pump room, water intrusion from a 1993 service water pipe break was found to have corroded several of the sensors. The instrument was replaced in March 1994. The inspectors observed that the instrument was in good condition and that the gasket seal was intact.

M8.3 (Closed) LER 50-440/94-016-00: "Overdue Surveillance Requirements Result in Noncompliance with Technical Specifications (TS)." On June 29, 1994, control room operators determined that one page was missing from the weekly TS rounds sheets 3-ring binder and that the surveillance requirements for emergency core cooling system (ECCS) actuation instrumentation channel checks had not been performed on the two previous shifts. An operator had removed some of the pages to repair torn pages and had not verified that all pages had been returned. At the time of the event, the plant was in cold shutdown. Because the sheet was missing, the ECCS instrument channel checks on that sheet were not performed. Several ECCS instrument channel checks exceeded TS out of service action statement allowable times by 8 hours and resulted in less than the TS required operable ECCS and emergency diesel generator systems. Associated TS action statements required suspension of operations with the potential for draining the reactor vessel. There had been no activities during this time with the potential for draining the vessel. Also, when the channel checks were performed, they were satisfactory and there was no evidence that any of the instruments had actually been inoperable. Therefore, the safety significance of this event was minimal. Corrective actions included coaching of the operators and additional training. The inspectors have monitored the TS rounds sheets over the past 2 years with no indications of additional problems in this area. This licensee-identified and corrected violation is being treated as a Non-Cited Violation (NCV 50-440/97004-01(DRP)), consistent with Section VII.B.1 of the NRC Enforcement Policy, NUREG-1600.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Facility Adherence to the Updated Safety Analysis Report (USAR)(37551)

While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the USAR that related to the areas inspected. The inspectors reviewed plant practices, procedures or parameters that were described in the USAR and documented the findings in this inspection report. For the USAR sections reviewed, no issues of plant configuration or USAR accuracy were identified. The licensee did identify a few minor issues which were entered in the licensee's corrective action program.

E4 Engineering Staff Knowledge and Performance

E4.1 High Pressure Core Spray Pump Operating Characteristics During Testing

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

The inspectors questioned engineering about an operability determination related to an emergency diesel generator (EDG) surveillance that was documented by PIF 97-0597.

b. Observations and Findings

As a result of questions from the NRC architect engineering inspection team concerning the speed droop setting of the Division III EDG, engineers discovered a possible inconsistency in SVI-E22-T1339, "Division III HPCS EDG 18 Month Loss Of Off-site Power (LOOP) Test." The SVI required the high pressure core spray (HPCS) pump to be operated at 4000 gallons per minute (gpm) while the LOCA analysis calculated the minimum requirement to be 6110 gpm at 518.4 pounds per square inch differential (psid). No design basis information was initially available to justify the difference.

Generic pump and motor curves provided by General Electric (GE) for the HPCS pump indicated that the motor amperage was highest at about 4000 gpm, which would be conservative for the LOOP test. However, the inspectors questioned the validity of the data in relation to the HPCS system, as installed. Specifically, the HPCS quarterly surveillance SVI-E22-T2001, "HPCS Pump and Valve Operability Test," indicated that at 6110 gpm at 518.3 psid the pump was in the high alert range, whereas the PIF indicated that 518.4 psid was the minimum. Also, the data provided by GE indicated that a 4000 gpm flow would correspond to approximately 970 psid. The LOOP SVI did not require pressure readings. The Perry staff was initially unable to explain the relational differences. Later the engineer obtained additional information that showed that data for the installed pump was consistent with the generic data. However, the additional information indicated that a slight change in the range of pump flow during the test would make it slightly more conservative. The motor power curve was relatively flat near the 4000 gpm point. The licensee planned to evaluate changing the SVI to incorporate the more conservative data.

c. Conclusion

The inspectors concluded that the Perry staff initially accepted the data from GE with insufficient questioning of the data's applicability to the Perry HPCS system. This was an example of a weakness in understanding of the design basis of the plant. However, the licensee did complete a more detailed review of the curves and determined that the surveillance test was acceptable.

IV. Plant Support

F5 Fire Protection Staff Training and Qualifications

a. Inspection Scope (71750)

The inspectors observed Periodic Test Instruction (PTI) P54-P0029, Rev. 2, "Fire Hydrant Semiannual Inspection" of outside Fire Hydrant No. 24.

b. Observations and Findings

The fire protection technician conducting the PTI was aware that the hydrant had a special function of providing fire protection water to cool the reactor should other sources fail. The technician explained the details of this function without prompting and described additional alternative methods of using the fire protection system to provide the reactor with cooling water. The technician also explained, in detail, the methods and bases for the PTI and the differences between Hydrant No. 24 and other hydrants.

c. Conclusion

The fire protection technician displayed awareness of the safety significance of the PTI he was performing and thorough knowledge of the fire protection system's capabilities to protect the reactor.

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 May 2, 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

J. P. Stetz, Senior Vice President, Nuclear
L. W. Myers, Vice President, Nuclear
R. D. Brandt, General Manager Nuclear Power Plant Department
W. R. Kanda, Director, Quality and Personnel Development Department
N. L. Bonner, Director, Nuclear Maintenance Department
J. J. Powers, Director, Nuclear Engineering Department
T. S. Rausch, Director, Nuclear Services Department
J. Messina, Operations Manager

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 60705:	Preparation for Refueling
IP 60710:	Refueling Activities
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71500:	Balance of Plant Inspection
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92720:	Corrective Action
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-440/97004-01	NCV	Missing page caused missed surveillance requirements

Closed

50-440/94001-00	LER	RWCU Isolation Due to Loss of Auxiliary Building Ventilation
50-440/94007-00	LER	Water Intrusion Leads to Passive Seismic Instrument Failure
50-440/94016-00	LER	Overdue Surveillance Requirements Result in Noncompliance with Technical Specifications
50-440/97003-00	LER	Loss of Electrical Power to Reactor Protection System Bus Due to Electrical Protective Assembly Trip Results in Engineered Safety Feature Actuation
50-440/97004-01	NCV	Missing page caused missed surveillance requirements

Discussed None

LIST OF ACRONYMS USED

APRM	AVERAGE POWER RANGE MONITOR
BOP	BALANCE OF PLANT
CFR	CODE OF FEDERAL REGULATIONS
CRD	CONTROL ROD DRIVE
CRER	CONTROL ROOM EMERGENCY RECIRCULATION
DRP	DIVISION OF REACTOR PROJECTS
EA	ENFORCEMENT ACTION
ECCS	EMERGENCY CORE COOLING SYSTEM
EDG	EMERGENCY DIESEL GENERATOR
EPA	ELECTRICAL PROTECTIVE ASSEMBLY
ERIS	EMERGENCY RESPONSE INFORMATION SYSTEM
ESW	EMERGENCY SERVICE WATER
FCV	FLOW CONTROL VALVE
FR	FEDERAL REGISTER
FW	FEEDWATER
GE	GENERAL ELECTRIC
GPM	GALLONS PER MINUTE
HPCS	HIGH PRESSURE CORE SPRAY
HPU	HYDRAULIC POWER UNIT
INPO	INSTITUTE OF NUCLEAR POWER OPERATIONS
LER	LICENSEE EVENT REPORT
LOCA	LOSS OF COOLANT ACCIDENT
LOOP	LOSS OF OFF-SITE POWER
LPCI	LOW PRESSURE COOLANT INJECTION
MOV	MOTOR-OPERATED VALVE
MSR	MOISTURE SEPARATOR REHEATERS
MSL	MAIN STEAM LINE
NPF	NUCLEAR POWER FACILITY
NRC	NUCLEAR REGULATORY COMMISSION
OEO	OPERATIONS EVOLUTION ORDER
PAP	PERRY ADMINISTRATIVE PROCEDURE
PDR	PUBLIC DOCUMENT ROOM
PIF	POTENTIAL ISSUE FORM
PORC	PLANT OPERATIONS REVIEW COMMITTEE
PSID	POUNDS PER SQUARE INCH DIFFERENTIAL
PTI	PERIODIC TEST INSTRUCTION
RCIC	REACTOR CORE ISOLATION COOLING
RHR	RESIDUAL HEAT REMOVAL
RPS	REACTOR PROTECTION SYSTEM
RSE	RESPONSIBLE SYSTEM ENGINEER
RWCU	REACTOR WATER CLEANUP
SCCM	STANDARD CUBIC CENTIMETERS PER MINUTE
SIL	SERVICE INFORMATION LETTER
SOI	SYSTEM OPERATING INSTRUCTION
STA	SHIFT TECHNICAL ADVISOR

SVI	SURVEILLANCE INSTRUCTION
TS	TECHNICAL SPECIFICATION
USAR	UPDATED SAFETY ANALYSIS REPORT