APPENDIX B

U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-313/87-30 Licenses: DPR-51
50-368/87-30 NPF-6

Dockets: 50-313
50-368

Licensee: Arkansas Power & Light Company (AP&L)
P. O. Box 551
Little Rock, Arkansas 72203

Facility Name: Arkansas Nuclear One (ANO), Units 1 and 2

Inspection At: ANO Site, Russellville, Arkansas

Inspection Conducted: September 1-30, 1987

Inspectors: for W. D. Johnson, Senior Resident Reactor Inspector — 10/15/87 Date

for C. C. Harbuck, Resident Reactor Inspector — 10/15/87 Date

Approved: J. P. Jaudon, Chief, Reactor Project Section A, Division of Reactor Projects — 10/19/87 Date

Inspection Summary

Inspection Conducted September 1-30, 1987 (Report 50-313/87-30)

Areas Inspected: Routine, unannounced inspection including operational safety verification, maintenance, surveillance, followup on previously identified items, followup on an allegation, and 10 CFR Part 21 reports.

Results: Within the six areas inspected, one apparent violation was identified (inadequate preventive maintenance program for lubricating pump couplings, paragraph 5).

Inspection Conducted September 1-30, 1987 (Report 50-368/87-30)

Areas Inspected: Routine, unannounced inspection of operational safety verification, maintenance, surveillance, followup on previously identified items, followup on an allegation, Technical Specification 4.8.1.1.2.c.12, and 10 CFR Part 21 reports.

Results: Within the seven areas inspected, one apparent violation was identified (inadequate preventive maintenance program for lubricating pump couplings, paragraph 5).

DETAILS

1. Persons Contacted

*J. Levine, Executive Director, Site Nuclear Operations
B. Baker, Operations Manager
E. Bickel, Health Physics Superintendent
J. Bruni, Shift Maintenance Supervisor
B. Cameron, Bechtel Lead Engineer
A. Cox, Unit 1 Operations Superintendent
E. Corliss, Instrumentation and Controls Supervisor
R. Douet, Quality Assurance Auditor
M. Durst, Project Engineering Superintendent
R. Dyer, Planning and Scheduling Coordinator
*E. Ewing, General Manager, Technical Support
B. Garrison, Operations Technical Support
D. Graham, Quality Control Engineering Supervisor
*H. Green, Quality Assurance Superintendent
L. Gulick, Unit 2 Operations Superintendent
C. Halbert, Engineering Supervisor
J. Hale, Bechtel Engineer
A. Hatley, Mechanical Maintenance Supervisor
H. Hollis, Security Superintendent
*D. Howard, Special Projects Manager
*L. Humphrey, General Manager, Nuclear Quality
G. Kendrick, Instrumentation and Controls Superintendent
J. Lamb, Safety and Fire Prevention Coordinator
*R. Lane, Engineering Manager
*D. Lomax, Plant Licensing Supervisor
A. McGregor, Engineering Services Supervisor
D. McKenney, Engineer
*J. McWilliams, Maintenance Manager
C. May, Acting Mechanical Maintenance Supervisor
*P. Michalk, Licensing Engineer
D. Payne, Maintenance Coordinator
*V. Pettus, Mechanical Maintenance Superintendent
S. Quennoz, General Manager, Plant Operations
P. Rehm, Mechanical Maintenance Engineering Technician
P. Rogers, Special Projects Coordinator
C. Shively, Plant Engineering Superintendent
*M. Smith, Reactor Engineering Supervisor
C. Taylor, Unit 2 Operations Technical Support Supervisor
*J. Taylor-Brown, Quality Control Superintendent
L. Taylor, Special Projects Coordinator
R. Wewers, Work Control Center Manager
M. White, Engineering Technician
B. Williams, Engineering Supervisor
T. Windham, Bechtel Engineer

G. Wrightam, Instrumentation and Controls Supervisor
C. Zimmerman, Unit 1 Operations Technical Support Supervisor

*Present at exit interview.

The NRC inspectors also contacted other plant personnel including operators, technicians, and administrative personnel.

2. Followup on Previously Identified Items (Units 1 and 2)

(Closed) Unresolved Item 368/8633-01: Failure to Meet the Original Acceptance Criteria for the Reference Control Element Assembly (CEA) Bank Reactivity Worth - The licensee resolved this deficiency, as discussed in NRC Inspection Report 50-313;368/86-33, by changing the acceptance criteria such that it encompassed the measured bank worth, with a temporary change to Procedure 2302.03, "Determination of CEA Worths by Exchange." The NRC inspector referred the supporting licensee safety evaluation to the NRC Office of Nuclear Reactor Regulation (NRR) for review. NRR concluded that additional justification should be provided by the licensee.

The following points formed the basis of the licensee's safety evaluation:

a. All of the test CEA bank reactivity worths were determined by measurement to be within their acceptance criteria.

b. Apparently a systematic deviation had affected either the measurement or prediction since all but one bank were more reactive than predicted.

c. One input to the process of determining the predicted bank worth was the bias applied to the computer calculated worth. (This bias will be further discussed below). For the Cycle 6 predicted worth determination, the bias had been increased over the value used in previous cycles. This had the effect of lowering the value of the predicted worths more than in past cycles.

d. The Final Safety Analysis Report (FSAR) applied a 10 percent uncertainty to the assumed worth of the most reactive CEA for both dropped and ejected CEA accident analysis. Thus, the actual worst case reactivity worth for each accident was still bounded by the FSAR analysis.

From these four points, the licensee concluded that the increased worth of the reference bank would neither increase the probability nor the consequences of any the accidents analyzed in the FSAR; additionally the other safety criteria of 10 CFR 50.59 would not be violated. However, for verification of this conclusion, the safety eval ation stipulated that further review be done following the completion of the 30 percent and 50 percent power physics testing.

The results of the staff's review of the safety evaluation were as follows:

The staff found that Item a. above was a valid point; especially since it was one of the three acceptance criteria for the Combustion Engineering rod swap method of measuring CEA bank worths discussed in Topical Report CEN-319-A. Further, it was noted that the second of the three acceptance criteria could have been mentioned since it was also met. This criteria was the total worth of all the CEAs. The one criteria not met was the measured reactivity worth of the reference bank.

Items b and c were considered to be related. Apparently the systematic deviation was the bias applied to the computer generated worth. This had affected the predicted values. The Cycle 6 calculated CEA bank worths had been determined by an NRC approved reactor core simulator computer code (ROCS). These results were then modified by the application of a bias expressed as a percentage of calculated worth. Generally, the bias is determined by statistically comparing the computer code generated results to historical measurement data. Application of the bias is meant to improve the accuracy of the predicted CEA bank worths. However, for Cycle 6, the fuel management core loading pattern was changed. Based on limited measurement data obtained from other Combustion Engineering reactor cores utilizing similar fuel management strategies and enrichments, the bias had been increased for Cycle 6. A recent reevaluation of this bias, using additional measurement data, now indicates that a lower value of bias is more appropriate (4 percent instead of 7 percent). Had this lower bias value been used for the Cycle 6 predictions, the reference bank worth acceptance criteria (±10 percent) would have been met.

The staff found that Item (d) contributed nothing to the basis of the safety evaluation because one cannot infer the effect of an increase in a CEA bank reactivity worth on the accident analysis by examining the analysis uncertainties in the assumed worth of an individual CEA in the FSAR accident analysis.

Additional support for the safety analysis was provided from examination of the core power distribution measurements at 30 percent, 50 percent, and 100 percent power. These measurements indicated that the portions of the core adjacent to the reference bank CEAs were slightly more reactive than predicted by Combustion Engineering. This would have the effect of increasing the reactivity worth of the reference bank.

The NRC staff pointed out two additional items which support the conclusion of the safety evaluation.

- Previous staff reviews of a number of rod swap methodologies indicate that the error which occurred on the reference bank (2 percent beyond the allowed plus/minus percent band) had little effect on the overall CEA bank measurement results.

- Examination of the measured critical positions with the estimated critical positions (and the critical boron concentration for the reference bank) for all the CEA banks showed good agreement except for one test bank. However, this test bank had a measured reactivity worth equal to the predicted value.

Based on the above, the NRC staff concluded that the higher than predicted reactivity worth of the reference bank posed no safety problem; that the licensee's initial safety evaluation, although not comprehensive in its basis, had reached an acceptable conclusion; and that no violation of NRC requirements had occurred.

However, the NRC staff pointed out that the Combustion Engineering Topical Report, CEN-319-A "Control Rod Group Exchange Technique," approved by the NRC, did not directly address the action to be taken should the reference bank measurement fall outside the acceptance criteria. The staff suggested that a better way to have resolved the problem in the Cycle 6 testing would have been to define the bank with the highest predicted reactivity worth as the new reference bank and measure its worth by boron dilution. The remaining test banks could then have been measured using this new reference bank by the rod swap method. This resolution method is allowed by the Topical Report.

The NRC inspector discussed this point with the licensee. The licensee committed to revise Procedure 2303.03 to include the contingency method noted above should the reference bank measurement fall outside the acceptance criteria in future cycle startup physics testing. The NRC inspector also suggested that NRR be consulted should activities in the future result in the desire to revise acceptance criteria which have been specifically approved by the NRC staff (such as the plus/minus 10 percent measured to predicted value of the reference bank stated in Topical Report CEN-319-A).

Based on the above, this item is closed.

(Closed) Unresolved Item 313/8718-02: <u>Basis for Process Monitor Calibration Procedure</u> - The NRC inspector reviewed the initial test procedure for the Unit 1 radiation monitoring system which was performed by the contractor, LFE Corporation, in 1971. The licensee's current calibration procedure is essentially equivalent to that procedure. Further, the licensee appeared to have an adequate understanding of the technical aspects of the calibration process. The NRC inspector concluded that the licensee has an adequate basis for the Unit 1 process monitor calibration procedure. This item is closed.

(Open) Unresolved Item 368/8525-01: <u>Process Radiation Monitor Calibration</u> - The principle concern of this item was the need to improve the measurement of the plateau region of the gaseous monitor Geiger-Muller detectors. Data from the recent calibration (June-August 1987) of a number of these monitors was reviewed to ascertain whether any improvements had been made. The procedure used, 2304.27, "Process Radiation Monitoring

System Calibration," Revision 14, was essentially unchanged from the previous revision used in the 1985 calibrations. Three problems were noted by the NRC inspector.

- It appeared that in several cases the plateau region extended beyond the upper limit of data collection, 1000 Vdc, set by the procedure. This item has been corrected as the new procedure (discussed below) requires measurements up to 1200 Vdc.

- What apparently was interpreted as the plateau region of Monitor 2RITS-8846, contained a peak. This had not been evaluated by the licensee.

- Monitor 2RITS-8845 high voltage had been adjusted to a value below the voltage range of the plateau during the calibration voltage adjustment.

Monitors 2RITS-8846 and -8845, which monitor the penetration room emergency ventilation exhaust, are not required by the Technical Specifications (TS). These two items were promptly addressed by the licensee. Using a new procedure (2304.06, "Gaseous Process Radiation Monitoring System Calibration," Revision 1, approved September 17, 1987) the monitors were recalibrated as part of Job Order 738245. The NRC inspector reviewed the data. The high voltage had previously been set around 850 to 900 Vdc.

This time both monitors were set at 1147 Vdc. At the time of the review, the licensee had reached no conclusion about the implications of this large adjustment in voltage. The licensee has plans to redo all the gaseous process monitors under the same job order in the near future due to problems with the old procedure. Further review to include the results of these additional recalibrations is needed before this item can be resolved. Therefore, this item remains open.

3. Operational Safety Verification (Units 1 and 2)

The NRC inspectors observed control room operations, reviewed applicable logs, and conducted discussions with control room operators. The NRC inspectors verified the operability of selected emergency systems, reviewed tagout records, and ensured that maintenance requests had been initiated for equipment in need of maintenance. The NRC inspectors made spot checks to verify that the physical security plan was being implemented. The NRC inspectors verified implementation of radiation protection controls during observation of plant activities.

The NRC inspectors toured accessible areas of the units to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibration. The NRC inspectors also observed plant housekeeping and cleanliness conditions during the tours.

The NRC inspectors walked down the accessible portions of the Unit 1 service water system to verify operability. The walkdown was conducted using Procedure 1104.29, Attachment "A", Revision 26, and Drawing M-210, Revision 47. No significant problems were noted.

During routine tours, the NRC inspector made the following observations:

- Stream Trap 2F-340 Isolation Valve 2MS-97-2 had a severe packing leak. The licensee was informed. Job Request 986631 was issued to stop the leak. This steam trap is in the line which drains the line just after Valve 2CV-1000-1, the "A" steam generator supply isolation to Emergency Feedwater Pump 2P7A's turbine driver, 2K3.

- The cable cover to 2LIT-4908 (Boric Acid Makeup Tank 2T6B level switch) was torn. This was repaired promptly by the licensee.

While touring the P36C makeup pump room the NRC inspector observed a seismic pipe support that did not conform to its design configuration. The hanger, HBD-20-SW-2, mounted vertically from the ceiling, supports the service water 3-inch diameter supply line to Room Cooler VUC-7C. One of the concrete expansion anchors had apparently broken free and had dropped almost completely out of its hole. After being informed, the licensee issued a Report of Abnormal Conditions (RAC 1-87-167) and Job Request 786933 to repair the hanger. The NRC inspector told the licensee that this particular noncompliance should be considered as another example of previous similar violations. The NRC inspector noted that a system walkdown program committed to by the licensee as long-term preventive corrective action for such violations, was just beginning. This program is designed to include identification of configuration control and degradation problems of safety-related piping systems including the seismic supports. The adequacy of this program will be evaluated during a future inspection.

The NRC inspector accompanied a health physics supervisor on a routine tour of the Unit 1 auxiliary building. It was noted that the contaminated controlled area around the Duratek filter equipment was in need of housekeeping attention. This problem was soon corrected. The NRC inspector noted no other significant problems, and considered the tour to have been conducted in a thorough manner.

These reviews and observations were conducted to verify that facility operations were in conformance with the requirements established under TSs, 10 CFR, and administrative procedures.

No violations or deviations were identified.

4. Monthly Surveillance Observation (Units 1 and 2)

The NRC inspector observed the TS required surveillance testing on Unit 2 Charging Pump 2P36A (Procedure 2104.02, Supplement I) and verified that testing was performed in accordance with adequate procedures, test instrumentation was calibrated, limiting conditions for operation were met, removal and restoration of the affected components were accomplished, test results conformed with TSs and procedure requirements, test results 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.

The NRC inspector also witnessed portions of the following test activities:

- Hydrogen sampler test (Procedure 1104.31, Supplement I)
- Test of motor-driven emergency feedwater pump after coupling inspection (Procedure 1106.06, Supplement I)
- Station battery pilot cell tests (Procedure 1307.16)
- Monthly test of emergency diesel generator (Procedure 1104.36, Supplement I)
- Test of emergency diesel generator to prove operability following failure of the other emergency diesel generator (Procedure 1104.36, Supplement II)
- Margin to saturation instrument channel calibration (Procedure 1304.84, Job Order 740691)
- Test of emergency diesel generator to prove operability following maintenance (Procedure 1104.36, Supplement II)
- Reactor building cooling coil service water flow test (Procedure 1104.33, Supplement VI)
- Monthly test of high pressure injection pump (Procedure 2104.39, Supplement III)
- Monthly test of low pressure injection pump (Procedure 2104.40, Supplement I)
- Calibration check of reactor building pressure instrument supplying Channel "C" of the reactor protection system (Procedure 1304.43)
- Monthly test of emergency diesel generator (Procedure 2104.36, Supplement II)

No violations or deviations were identified.

5. Monthly Maintenance Observation (Units 1 and 2)

Station maintenance activities of safety-related systems and components listed below were observed to ascertain that they were conducted in accordance with approved procedures, Regulatory Guides, and industry codes or standards; and in conformance with TSs.

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

The following maintenance activities were observed:

- Replace Auxiliary Relay 152-408/X (Job Order 739521)
- Replace motor to gear coupling on P36C (Job Order 739489, Procedure 1402.010)
- Replace leaking discharge flange gasket on high pressure safety injection pump (Job Order 737405)
- Replace starting air pressure regulators on emergency diesel generator (Job Order 735623)
- Replace air start solenoid valves on emergency diesel generator (Job Orders 727428 and 735454)
- Diesel fire pump quarterly surveillance inspection (Procedure 1306.27, Job Order 738311)
- Inspection of pump end coupling of Charging Pump 2P36A (Job Order 740070)
- Inspection of high pressure injection pump coupling (Job Order 740070, Procedure 2402.36)

On September 1, 1987, the NRC inspector observed the disassembly of the motor to gear coupling on the "C" makeup pump on Unit 1. It was suspected that this coupling had failed since operators had observed the motor running and the pump not rotating after the pump lost discharge pressure.

The coupling disassembly and inspection were performed under Job Order 739489. The coupling gears were found to be badly worn, and only a small amount of hardened grease was found in the coupling. Several days later, after obtaining a new coupling and revising Procedure 1402.010 to provide maintenance guidance, the coupling was replaced. Under the same job order, the NRC inspector observed the inspection of the gear to pump coupling on the "C" makeup pump. This coupling and its grease were found to be in good condition.

The NRC inspector reviewed the technical manual for the makeup pumps, Byron Jackson Technical Manual G404550, and discussed with licensee personnel how the motor to gear coupling limited the motor end float. It was determined that buttons or spacer discs are not needed to limit end float due to the clearances established when the gear and motor were mounted on the base plate.

Step 7.18.2 of Procedure 1402.010 includes instructions to install spacer plates and buttons in the motor to gear coupling. Since these are not needed and not used, licensee personnel stated that this section of the procedure would be revised. In addition, several other sections of this procedure will be revised to incorporate the temporary changes which were made prior to installation of the new motor to gear coupling. This item will remain open pending revision of Procedure 1402.010 and NRC inspector review of the revised procedure. (313/8730-02)

During the followup of the P36C motor coupling failure event, the NRC inspectors determined that the licensee's preventive maintenance program for safety-related pump couplings was deficient, in that the lubrication schedules were not well defined and apparently not followed. This conclusion was based on the licensee being unable to find lubrication records of several pump couplings subsequent to the date indicated below:

. High Pressure Injection (HPI) Pump P36A motor coupling	No record
. HPI Pump P36C pump coupling	July 1979
. HPI Pump P36C motor coupling	1982
. Low Pressure Injection (LPI) Pump P34A	May 1982
. LPI Pump P34B	No record
. Reactor Building Spray (RBS) Pump P35A	April 1979
. RBS Pump P35B	No record
. Emergency Feedwater Pump 2P7B	No record

Inadequate lubrication was apparently the primary contributor to the P36C motor coupling failure.

Lubrication schedules for safety-related equipment are specified in Regulatory Guide 1.33, Appendix "A", which is committed to by TS 6.8.1 of both units. Failure to establish and implement adequate lubrication preventive maintenance schedules is an apparent violation. (313;368/8730-01)

The NRC inspectors noted that the licensee pursued an aggressive program to identify and correct possible generic problems indicated by the P36C coupling failure. This program included:

a. Inspection of all high speed lubricated couplings on safety-related pumps (and also on nonsafety-related pumps). This effort was completed for safety-related pumps on September 14, 1987. No additional lubrication problems were noted.

b. Determination of the status of the safety-related preventive maintenance program. This effort was scheduled to be completed by October 9, 1987. Overdue PMs identified will be scheduled as appropriate.

The NRC inspectors will continue to follow the licensee's implementation of this program. The licensee was informed that the generic implications of the P36C coupling failure, already being pursued as just noted, and the corrective actions taken as a result, should be addressed in the response to the violation noted above.

6. Followup of Allegation 4-86-A-042 (Units 1 and 2)

The following concern obtained from the review of statements provided to the NRC, was addressed during this inspection.

An alleger stated that at unspecified times and locations unqualified workers used sharp stainless steel hooks to remove foam from electrical conduits without deenergizing the cables in the conduits. Discussions with site personnel indicated that poor practice in this area had been a problem in the past. The NRC inspector reviewed a memo dated July 18, 1984, which stated a policy that no sharp metal tools would be used for penetrating and removing foam. The NRC inspector reviewed the file on a personnel injury accident which occurred on October 24, 1984. In this case, an engineer was injured when using a nylon probe with a metal hook attached to one end to check the depth of a foam dam installed in a conduit under a motor control center. When the probe was removed, the metal contacted the feeder bus in the motor control center causing a short resulting in a fireball and severe personnel injury. Following this incident, a policy of requiring an approved work plan for penetration sealing work in energized electrical equipment was adopted. This policy required consideration of physical barriers and electrical isolation.

The NRC inspector reviewed Procedure 4033.06, "Installation, Repair and/or Alteration of Fire Barrier Penetration Seals." This procedure requires that qualified personnel perform penetration seal work and includes requirements that only nonconducting instruments shall be used when working with electrical equipment and that no sharp or metal tools shall be used in removing old sealants from around cables.

The NRC inspector concluded that the allegation was probably correct and that the poor practice resulted in a personnel injury in October 1984.

Since that time, controls over penetration sealing have been improved and they appeared to be adequate at the time of this inspection.

This allegation remains open pending the review of the other technical concerns identified during the review of the statements provided to the NRC.

No violations or deviations were identified.

7. Interpretation of TS Surveillance Requirement 4.8.1.1.2.c.12 (Unit 2)

TS 4.8.1.1.2.c.12 states, "Each diesel generator shall be demonstrated OPERABLE: At least once per 18 months during shutdown by: Verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by
(1) returning the diesel generator to standby operation and
(2) automatically energizes the emergency loads with offsite power."

During the last performance of this surveillance, the NRC inspector had commented that the licensee's procedure did not appear to meet the intent of this TS in that no loads were automatically started as described in (2) above; but rather the actuation of relays in the breaker closing circuits for the associated emergency loads were observed and timed. At that time, the licensee revised their procedure to require the actual starting of some, but not all, of the loads possible. This was considered an acceptable alternative at the time by the NRC inspector.

During this inspection period, preparation of the surveillance test procedure to be used in the upcoming 1988 refueling outage was in progress, and the licensee requested additional technical guidance on the correct interpretation of this TS, from the NRC inspector.

A conference call between the licensee, the NRR project manager, an NRR technical staff representative, and the NRC inspector, was held on September 22, 1987, to discuss this TS. The licensee was told by NRR that the proper interpretation of Item (2) of the TS was to automatically energize all the emergency loads that were possible with offsite power. It was suggested that a change of this interpretation could be pursued by means of a TS amendment proposal.

Persons participating in the conference call were:

NRR	AP&L	NRC Site
Jim Knight	Larry Taylor	Craig Harbuck
George Dick	Don Lomax	

No violations or deviations were identified.

8. Part 21 Reports (Units 1 and 2)

During review of 10 CFR Part 21 reports submitted by manufacturers, suppliers, and reactor licensees, NRC personnel identified certain reports which could be applicable at ANO. The NRC inspector provided copies of these reports to the licensee. The reports provided are listed in the attached table. It is expected that the licensee will review these reports to determine whether they are applicable to equipment at ANO.

Part 21 Reports Provided to Licensee

Number	Originator	Date	Subject
86-002	Georgia Power Co.	09/05/86	Pipe support tolerance and installation procedures
86-003	Indiana & Michigan Electric Co.	09/18/86	Defective emergency head lever supplied for aux feed pump
86-009	Georgia Power Co.	07/31/86	Spring failure-Valcor solenoid valves
86-013	Foxboro Co.	10/07/86	Advisory on handling Foxboro N-E11 and N-E13 transmitters
87-002	Virginia Electric and Power Co.	11/12/86	Potential defect in new svc water spray support system
87-003	Foxboro Co.	06/04/86	End of Life susceptibility of e-line & H-line instruments
87-004	Indiana & Michigan Electric Co.	12/20/85	Weld electrodes with incomplete flux coating
87-005	Florida Power & Light Co.	07/08/86	Tip damage on anti-reverse rotation device pins on RCP
87-006	Portland General Electric Co.	07/25/86	Stationary sleeve on MSIV thrust bearings interference
87-007	PROMATEC, Inc.	02/17/86	Defective conduit seals
87-011	General Electric Co.	11/17/86	HFA relays could experience incorrect operations
87-016	Limitorque Corp.	12/19/86	Damaged insulation on Limitorque valve operator DC motor

87-019	Vermont Yankee Nuclear Power Corp.	11/10/86	Design defect in Limitorque valve operators pre-1975
87-020	Automatic Sprinkler Corp.	12/01/86	Automatic fire sprinkler system valve failure
87-025	GA Technologies, Inc.	02/23/87	Low insulation resistance of coax cable for HRR monitors
87-028	Niagara Mohawk	01/26/87	Improper seating of Agastat GP series relays
87-029	Toledo Edison	02/03/87	Inadequate instructions to maintain torque switch balance
87-030	Niagara Mohawk	02/02/87	Improper electrical duct seal design
87-031	Automatic Valve Corp.	12/19/86	Houghto 620 lubricant attacks & degrades aluminum valves
87-035	Foxboro Co.	02/17/87	Foxboro spec 200 C/V cards affected by high moisture
87-036	Sacramento Municipal Utility District	02/10/87	Limitorque warped limit switch rotors
87-038	Morrison-Knudsen Co., Inc.	01/13/87	EDG control relay failed to drop out when deenergized
87-044	Arizona Nuclear Power Project	03/02/87	Replacement fuel injection tube nuts not per SAE J521b
87-046	Isomedix	03/30/87	EQ qualification questionable

9. Exit Interview

The NRC inspectors met with Mr. J. M. Levine, Executive Director, ANO Site Operations, and other members of the AP&L staff at the end of the inspection. At this meeting, the NRC inspectors summarized the scope of the inspection and the findings.

ATTACHMENT 2

UPDATE FORM

FACILITY: ~~[illegible]~~ ANO-2
DOCKET: 50-~~313~~ 368

ORIGINATORS NAME: HARBUCK/JOHNSON

TYPE: V

ITEM NO.: 8730-01

REPORT: 8730

PARAGRAPH: 5

FUNCTIONAL AREA: MAINTENANCE

DESCRIPTION: INADEQUATE PM LUBRIC
ATION SCHEDULES FOR
PUMP COUPLINGS

STATUS CODE: O

UPDATE/CLOSE:
REPORT

RESPONSIBLE
SECTION RPSB

DETAILS:

ATTACHMENT 2

UPDATE FORM

FACILITY: ~~50~~ ANO-1
DOCKET: 50-313

ORIGINATORS NAME: HARBUCK/JOHNSON

TYPE: V

ITEM NO.: 8730-01

REPORT: 8730

PARAGRAPH: 5

FUNCTIONAL AREA: MAINTENANCE

DESCRIPTION: INADEQUATE PM LUBRICATION SCHEDULES FOR PUMP COUPLINGS

STATUS CODE: O

UPDATE/CLOSE: REPORT RPSB

RESPONSIBLE SECTION

DETAILS:

ATTACHMENT 2

UPDATE FORM

FACILITY: ANO-1
DOCKET: 50-313

ORIGINATORS NAME: JOHNSON W D

TYPE: 0

ITEM NO.: 8730-02

REPORT: 8730

PARAGRAPH: 5

FUNCTIONAL AREA: MAINTENANCE

DESCRIPTION: REVIEW REVISED

MAKEUP PUMP MAINT

PROC 1402.010

STATUS CODE: 0

UPDATE/CLOSE:
REPORT

RESPONSIBLE
SECTION RPSB

DETAILS:

Inspector Name	Type	Item No.	Report Paragraph	Functional Area	Description	Resp. Sect.	(Update)/(Close) Report	ST
Martin, T.	U	8601-006	86-01 III.A.7	Design Change	Design Drawing Errors	RPSB		O
Martin, T.	U	8601-009	86-01 III.B.3	Maintenance	T7A Maintenance and Testing	RPSB		O
Harbuck, C.	U	8718-002	87-18 4B	Surveillance	Basis Needed For Process Monitor Calibration	RPSB	8730	C
Johnson, W.	U	8718-003	87-18 4D	Surveillance	Penetration Room Ventilation Sys Performance Demo	RPSB		O
Harrell, P.	U	8721-001	87-21	Training	Decrease in Scope of Requal Prog Without Prior NRC Approval	RPSB		O
Harrell, P.	U	8721-003	87-21	Training	Failure to Perform An Annual Evaluation of All Licensed Oper	RPSB		O
Kelly, J.	V	8323-001	5		Security-Related Equipment not maint. in operable condit.	RP&SB		O
Kelly, J.	V	8323-002	5		Security-related equipment not maint. in operable condit.	RP&SB		O
Kelly, J.	V	8323-003	11		Security plan not available	RP&SB		O
Kelly, J.	V	8512-002	7		Inadequate Security System Management	RP&SB		O
Johnson, W.	V	8527-001	85-27 2	Plant Operations	Control Room Ventil- Procedure Inadequate	RPSB	(87-05)(8718)(8718)	O

Inspector Name	Type	Item No.	Report Paragraph	Functional Area	Description	Resp. Sect.	(Update)/(Close) Report	ST
Johnson, W.	U	8525-001	5		Process Radiation Monitor Calibration	RPSB	(8730)	0
Harbuck, C.	U	8633-001	86-33 5	Outage	Physics Test Acceptance Criteria Change	RPSB	8730	~~0~~ C
Harrell, P.	U	8721-001	87-21	Training	Decrease in Scope of Requal Prog Without Prior NRC Approval	RPSB		0
Harrell, P.	U	8721-003	87-21	Training	Failure to Perform An Annual Evaluation of All Licensed Oper	RPSB		0
Kelly, J.	V	8512-001	5		Inadequate Managemnt Control Impacting Security	RP&SB		0
Kelly, J.	V	8512-002	7		Inadequate Security System Maintenance	RP&SB		0
Johnson, W.	V	8523-001	4		Failure maintain fire doors shut	RPSB		0
Caldwell, R.	V	8606-001	86-06 2	Safeguards		RP&SB		0
Caldwell, R.	V	8606-002	86-06 3	Safeguards		RP&SB		0
Stewart, R.	V	8615-001	86-15 11	Quality Programs	Calibration of Admixture Dispenser	OS		0
Johnson, W.	V	8615-008	86-15 4	Maintenance	Control over scaffold erection near safety related equip	RPSB		0