APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-313/86-21
50-368/86-22

Licenses: DPR-51
NPF-6

Dockets: 50-313
50-368

Licensee: Arkansas Power and 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: June 1-30, 1986

Inspectors: W. E. Vaughn for W. D. Johnson, Senior Resident Reactor Inspector (pars. 2, 3, 4, 5, 6, 7) — Date 7/15/86

W. E. Vaughn for C. C. Harbuck, Resident Reactor Inspector (pars. 4, 5, 6, 10) — Date 7/15/86

W. E. Vaughn for G. S. Vissing, Project Manager (pars. 8, 9) — Date 7/15/86

Approved: D. R. Hunter, Chief, Reactor Projects Section B, Reactor Projects Branch — Date 7/16/86

Inspection Summary

Inspection Conducted June 1-30, 1986 (Report 50-313/86-21)

Areas Inspected: Routine, unannounced inspection including operational safety verification, maintenance, surveillance, followup on previously identified items, followup on Licensee Event Reports, followup on IE Bulletin 85-03, review of transient reports, and 10 CFR 50.59 evaluations.

Results: Within the eight areas inspected, one violation was identified (failure to document safety evaluations as required by 10 CFR 50.59, paragraph 9).

Inspection Summary

Inspection Conducted June 1-30, 1986 (Report 50-368/86-22)

Areas Inspected: Routine, unannounced inspection including operational safety verification, maintenance, surveillance, followup on Licensee Event Reports, followup on IE Bulletin 85-03, 10 CFR 50.59 evaluations, and refueling activities.

Results: Within the seven areas inspected, one violation was identified (failure to follow post-maintenance test procedure, paragraph 6).

DETAILS

1. Persons Contacted

J. Levine, Director of Site Nuclear Operations
R. Ashcraft, Electrical Maintenance Supervisor
*B. Baker, Operations Manager
R. Bennett, Consultant, Wyle Laboratories
J. Brown, Quality Assurance Supervisor
*P. Campbell, Licensing Engineer
*M. Cooper, Quality Assurance Inspector
A. Cox, Operations Technical Support Supervisor
*E. Ewing, General Manager Technical Support
M. Goodson, Civil Engineer
L. Gulick, Unit 2 Operations Superintendent
H. Hollis, Security Coordinator
D. Horton, Quality Assurance Manager
*D. Howard, Special Projects Manager
L. Humphrey, General Manager, Nuclear Quality
D. Johnson, Licensing Engineer
H. Jones, Field Construction Manager
P. Kearney, Project Engineer
*R. Lane, Engineering Manager
*D. Lomax, Licensing Supervisor
B. Lovett, Electrical Maintenance Engineer
A. McGregor, Engineering Services Supervisor
*J. McWilliams, Maintenance Manager
V. Pettus, Mechanical Maintenance Superintendent
D. Provencher, Quality Engineering Supervisor
E. Rice, Electrical Maintenance Supervisor
P. Rogers, Plant Licensing Engineer
C. Shively, Plant Engineering Superintendent
R. Simmons, Planning and Scheduling Supervisor
C. Taylor, Operations Technical Support Supervisor
R. Taylor, Consultant, Wyle Laboratories
B. Terwilliger, Operations Assessment Supervisor
D. Wagner, Health Physics Supervisor
*R. Wewers, Work Control Center Manager
C. Zimmerman, Operations Technical Support

*Present at exit interview.

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

2. Followup on Previously Identified Items (Unit 1)

(Closed) Deviation 313/8523-01: Failure to meet a commitment relative to implementation of B&W Safe-End Task Force recommendations.

The licensee's response to this item included a clarification of the commitments made in their letter to the NRC dated April 22, 1985. The licensee provided this clarification and commitment revision to the NRC Office of Nuclear Reactor Regulation in a letter dated March 10, 1986.

3. Licensee Event Report (LER) Followup (Units 1 and 2)

Through direct observation, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence has been accomplished in accordance with Technical Specifications.

Unit 1

86-004, Reactor Trip

Unit 2

85-024, Fire Barrier Seal Degradation
86-001, Reactor Trip

Unit 1 LER 86-004 reported a reactor trip on high reactor coolant system pressure due to a turbine control system malfunction. The root cause was believed to be a lightning strike which disturbed the turbine control system through the plant grounding system. The NRC inspector reviewed Report of Abnormal Conditions (RAC) 1-86-070 and the job order used to troubleshoot, repair, and test the turbine control system.

Unit 2 LER 85-024 reported degradation of the fire barrier seal between the auxiliary building floors and the reactor building wall. This barrier was sealed and added to the fire barrier surveillance program.

Unit 2 LER 86-001 reported a reactor trip due to a spurious closure of a main steam isolation valve. The NRC inspector reviewed RAC 2-86-018 and the job orders used to replace the relays suspected of causing the valve closure. (JO 708120 and 708228)

No violations or deviations were identified.

4. 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 inspectors verified the operability of selected emergency systems, reviewed tagout records, verified proper return to service of affected components, and ensured that maintenance requests had been initiated for equipment in need of maintenance. The inspectors made spot checks to verify that the physical security plan was being implemented in accordance with the station security plan. The inspectors verified implementation of radiation protection controls during observation of plant activities.

The NRC inspectors toured accessible areas of the units, including the Unit 2 containment building, to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibration. The inspectors also observed plant housekeeping and cleanliness conditions during the tours.

During a plant tour, the NRC inspector noted that pipe hanger 2DCB-3-H9 on emergency feedwater piping, downstream of 2FE-0717 and upstream of 2CV-1075-1, was carrying no load. The hanger drawing for this hanger indicated a dead weight load of 197 pounds downward. This condition was identified to the licensee, and the hanger was adjusted.

While touring the Unit 2 reactor building, the NRC inspector found deficiencies in several seismic class I pipe supports. These deficiencies were subsequently documented by the licensee under two RACs (Report of Abnormal Conditions) as follows:

(1) RAC 2-86-121 'A' steam generator

- loose baseplate for pipe hanger supporting line 2DBB-1000-3/4" containing 2SGS-21A, a level transmitter lower tap isolation valve.

- loose anchor bolts on pipe support 2DBB-7-H5 which supports line 2DBB-7-4" the steam generator blowdown line.

(2) RAC 2-86-122 'B' reactor coolant pump

- loose anchor bolts on the two pipe supports adjacent to each side of 2RCP-3B, the controlled bleedoff manual isolation valve.

The licensee issued Job Requests 15488 and 15489 to correct these deficiencies and planned to perform an engineering evaluation of their effect on operability of the supports. These deficiencies will remain as an open item pending review of the licensee repair of the supports and of the completed operability evaluation. (368/8622-01)

The NRC inspector observed that two of the Unit 1 service water pump discharge pipes were corroded, as were the discharge strainer body for P4C and the piping flange bolts for P4A. A licensee representative stated that a job order had been issued to remove the corrosion and paint the affected areas.

The NRC inspectors walked down the accessible portions of the Unit 1 and Unit 2 emergency boration flowpaths. The walkdown was performed using Procedures 1104.03, 2104.03, 1202.01, and 2202.01 and Drawings M-231, M-233, and M-2231. No system lineup discrepancies were identified but minor errors were noted on Drawing M-233. These were identified to the licensee for correction.

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

No violations or deviations were identified.

5. Monthly Surveillance Observation (Units 1 and 2)

The NRC inspector observed that the Technical Specification required surveillance testing on the systems listed below 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 Technical Specifications 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 inspector witnessed portions of the following test activities:

- Monthly channel test of reactor protection system channel A (Procedure 1304.37) (JO 712931)
- Engineered safeguards actuation system analog channel number 1 test (Procedure 1304.49) (JO 713104)
- Escape lock barrel test (JO 712934)
- Hydrogen purge system test (Procedure 1104.33, Supplement II)
- Service water flow test (Procedure 2311.02)
- Test of control room emergency ventilation unit VSF-9 (Procedure 1104.34, Supplement 1A)
- Monthly channel test of reactor protection system channel C (Procedure 1304.39) (JO 713858)
- Station battery 2D12 service discharge test (Procedure 2403.26) (JO 708158)
- 18-month operability test of the 'B' diesel generator (Procedure 2104.36, Supplement 4)
- Overspeed test of the 'B' diesel generator (Procedure 2104.36, Supplement 7)
- EFW pump P7A monthly test (Procedure 1106.06, Supplement 2)

No violations or deviations were identified.

6. 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 Technical Specifications.

The following items were considered during this review: the limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; and fire prevention controls were implemented.

Work requests were reviewed to determine 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:

- Service water pump preventive maintenance (Procedure 2402.34) (JO 711288)
- Repair Target Rock solenoid valve position indication (Procedure 1403.02) (JO 713486)
- Diesel generator fuel solenoid valve inspection (JO 0233)
- 18-month inspection of diesel generator (Procedure 2306.05) (JO 707885)
- C high pressure injection pump motor preventive maintenance (JO 525514)
- Condensate storage tank piping (Design Change Package 82-2086B) (JO 703886)
- Condensate storage tank foundation and pipe chase (DCP 82-2086)
- Uncouple motor from C high pressure injection pump (JO 711764)
- Anchor bolt replacement (DCP 84-2043) (JO 711454)
- Battery charger 2D34 load test following maintenance (Procedure 2403.53) (JO 714504)

The NRC inspector found that the lineup specified in Procedure 2403.53 for testing battery charger 2D34 was not being used during the

post-maintenance test. The procedure lineup requires that 2D34 be loaded by bus 2D01 or 2D02 and that a testing unit be used to obtain a load of 200 amps on the battery charger. During the test observed on June 27, 1986, battery charger 2D34 was not connected to either battery bus, and the testing unit was supplying the total load. Breaker 72-0211, which was specified to be closed by the procedure, was tagged open; and the green power leads from 2D34 to 2D02, which are not addressed in the procedure, were lifted. No procedure change to permit this lineup had been obtained. This is an apparent violation. (368/8622-02)

7. Followup on IE Bulletin (IEB) 85-03 (Units 1 and 2)

IEB 85-03 was issued on November 15, 1985, and is entitled, "Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings." The licensee's initial response to the bulletin was dated May 14, 1986. The NRC Office of Inspection and Enforcement is conducting a technical review of this response.

The NRC inspector reviewed the licensee's program for complying with IEB 85-03. This program is contained in Work Plan 1409.56 and coordinates IEB 85-03 activities with other motor-operated valve activities such as environmental qualification (EQ) inspections and preventive maintenance. The NRC inspector also reviewed the differential pressure testing procedure (1409.14), the MOVATS testing procedure (1403.31), and the EQ inspection procedures (1407.37 and 1407.38).

MOVATS testing was observed for 2CV-5128, 1025, 0711, and 0789. EQ inspections were observed for 2CV-1511 and 1039. These activities were found to be conducted in accordance with the approved procedures. The NRC inspector will observe more testing and inspection activities on motor-operated valves during the next inspection period.

No violations or deviations were identified.

8. Review of Transient Reports (Unit 1)

ANO-1 had 12 transients in 1985. Six transients were automatic trips, two were manual trips soon after the events which required plant shutdown, and eleven transients were the result of malfunctions in the main feedwater system. All transient reports were reviewed. The reports reviewed were unofficial copies. The formal reports which were on microfilm were not reviewed. However, the inspector was assured that the reports which were reviewed were, for the most part, the same as the formal reports. The following observations were noted:

a. Most reports of transients resulting in plant trips had good plots of plant parameters.

b. One report had a typed page of the sequence of events. Most reports had a computer printout of the sequence of events. One report did not have a sequence of events.

c. Two events involved manual initiation of HPI. This is considered significant but was not discussed in the description of the event.

d. To control the transients to preclude an automatic trip, the pilot operated relief valve (PORV) and pressurizer spray were operated during some events. This is considered significant but was not always discussed in the description of the event.

e. It was difficult to identify the plant parameters in the safety parameter display system (SPDS) computer printouts because the parameters were not identified by name. Also, the SPDS plotted data was difficult to read.

No violations or deviations were identified.

9. Review of the 10 CFR 50.59 Determinations of Procedure Changes and Design Change Packages (Units 1 and 2)

The procedure for Procedure Review, Approval and Revision Control, 1000.06, Revision 22 was reviewed as it applied to the requirements for the 50.59 determination. The procedure provided for a 50.59 review and determination by completion of a one page form 1000.06E consisting of seven questions with boxes of "Yes" or "No" to check. The seven questions did relate to the three factors of the determination for an unreviewed safety question as required by 10 CFR 50.59. There was a small space for a narrative discussion for the basis for the determination. There were no instructions for completing the form. Also, there was no indication for the need for a written safety evaluation for the basis of the determination.

The procedure for Design Control, 1032.01, Revision 7 was reviewed. Attachment 7 of the procedure provided guidelines for preparation of a safety and environmental determination which included a 50.59 determination. The guidance appeared to be very complete; however, the determination is recorded on a Form 202F9 which provided for a check off for the three factors for determining that an unreviewed safety question was not involved. The form provided for a brief justification for the "No" answers given in check off. It also provided for the identification of the FSAR sections involved. Although the guidance provided a discussion for the need of a written safety evaluation, the Form 202F9 provided no space specifically for a safety evaluation. It is implied then that the Form 202F9 is the Safety Evaluation.

The procedure and a change to the procedure for Motor Driven Emergency Feedwater Pump Test, 1311.04, Revision 0 was reviewed. The change package provided a Form 1000.06E. All seven boxes to the questions were checked off as "No." The "Basis of Determination" included a brief description of how and why the test was to be performed. The determination was not explicit on why the procedure or change was not an unreviewed safety

question. The three factors for making the 50.59 determination were not discussed and there was no written safety evaluation for the determination.

The following design change packages (DCPs) were reviewed for the 50.59 determinations:

a. DCP 81-1080A, Modification of Diesel Generator Voltage Shutdown Device -- Electrical

b. DCP 83-1007, Provide Disconnect Switch for "B" Service Water Pump

c. DCP 83-1012, Provide Isolation Device for Pressurizer ERV Block Valve

d. DCP 83-1107, Acoustic Valve Monitoring System Replacement

e. DCP 83-1170, LPI Flow Indication, HPI Flow Indication, Containment Spray Flow

For each DCP Form 202F9 was completed with a brief justification for the "No" checks (answers to questions which address the three factors which must be considered in the determination if the change is an unreviewed safety question). There were no written safety evaluations which provided bases for the determinations. Also, in the "Justifications," each of the three factors necessary for the determination that there were no unreviewed safety questions were not explicitly addressed. Revisions to the DCPs did not have 50.59 determinations.

Failure to provide written safety evaluations which provide the bases for the determination that the above test and design changes did not involve an unreviewed safety question is an apparent violation. (313/8621-01)

The DCPs did have very detailed descriptions of the changes involved. These descriptions could be good foundations to 50.59 determinations, but were not used for this purpose. The Form 202F9 did address the Safety Analysis Report sections affected.

The licensee recognizes weaknesses in the 50.59 determinations and has started a program to improve this area. A policy paper is under preparation which addresses the training and guidance which, for the most part, is provided in Attachment 7 of the Procedure for Design Control.

10. Refueling Activities (Unit 2)

The purpose of this area of inspection was to ascertain whether refueling activities are being controlled and conducted as required by Technical Specifications (TS) and approved procedures.

The NRC inspector reviewed the following licensee procedures and compared them to Section 9 of the TS. It appeared that the procedures implemented the TS properly.

2502.01	"Refueling Shuffle"
2502.03	"Preparation for Refueling"
1015.03B-8	"Mode 6 Log"

Through routine daily observations and specific observations of fuel movement during core off load from both inside the reactor building and the spent fuel area of the auxiliary building, the NRC inspector concluded that the licensee appeared to be conducting refueling activities in accordance with the approved procedures.

No violations or deviations were identified.

11. Exit Interview

The NRC inspectors met with Mr. E. Ewing, Acting Director, Site Nuclear Operations, and other members of the AP&L staff at the end of this inspection. At this meeting, the inspectors summarized the scope of the inspection and the findings.