APPENDIX C

U. S. NUCLEAR REGULATORY COMMISSION

REGION IV

Report: 50-313/82-05 Licenses: DPR-51
50-368/82-05 NPF-6

Dockets: 50-313
50-368

Licensee: Arkansas Power and Light Company

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

Inspection At: ANO Site, Russellville, Arkansas

Inspection Conducted: March 1-31, 1982

Inspectors: W. D. Johnson 4/5/82
W. D. Johnson, Senior Resident Reactor Inspector Date
(Paragraphs 1, 2, 2a, 2b, 3, 4, 5, 6, 7)

L. J. Callan 4/6/82
L. J. Callan, Resident Reactor Inspector Date
(Paragraphs 1, 2, 2c, 3, 4, 8)

Approved: R. E. Hall 4/12/82
R. E. Hall, Chief, Reactor Project Section C Date

Inspection Summary

Inspection conducted during period of March 1-31, 1982 (Report 50-313/82-05)

Areas Inspected: Routine, announced inspection including operational safety verification, surveillance, maintenance, follow up on IE Bulletin 80-06, and review of Licensee Event Reports.

The inspection involved 69 inspector-hours on site by two NRC inspectors.

Results: Within the five areas inspected, one apparent violation was identified in one area (two of three high pressure injection pumps inoperable as reported in Licensee Event Report 313/82-003/03L-0, paragraph 7).

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Inspection conducted during period of March 1-31, 1982 (Report 50-368/82-05)

Areas Inspected: Routine, announced inspection including operational safety verification, surveillance, maintenance, follow up on IE Bulletin 80-06, and follow up on plant trip.

The inspection involved 81 inspector-hours on site by two NRC inspectors.

Results: Within the five areas inspected, one apparent violation was identified (reactor protection channel not tripped when required, paragraph 2a) and one apparent deviation was identified (testing per IE Bulletin 80-06 not performed as committed, paragraph 5).

DETAILS SECTION

1. Persons Contacted

   J. P. O'Hanlon, ANO General Manager
   J. Levine, Engineering & Technical Support Manager
   B. A. Baker, Operations Manager
   T. N. Cogburn, Plant Analysis Superintendent
   E. C. Ewing, Plant Engineering Superintendent
   L. Sanders, Maintenance Manager
   J. McWilliams, Unit 1 Operations Superintendent
   J. Albers, Planning and Scheduling Supervisor
   M. J. Bolanis, Health Physics Superintendent
   R. Wewers, Unit 2 Operations Superintendent
   D. Wagner, Health Physics Supervisor
   M. Stroud, Production Engineer
   L. Dugger, Special Projects Manager
   L. Humphrey, Administrative Manager
   R. Turner, Electrical Engineering Supervisor
   J. Roberson, I&C Supervisor
   J. Benham, I&C Supervisor
   M. Asher, Planning and Scheduling Coordinator
   M. Konya, Nuclear Engineer
   J. Marshall, Licensing Manager
   R. Tucker, Electrical Maintenance Superintendent
   B. Scalco, Electrical Maintenance Supervisor

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

2. 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, and verified proper return-to-service of affected components. Tours of accessible areas of the units were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibration, and to verify that maintenance requests had been initiated for equipment in need of maintenance. The inspectors, by observation and direct interview, verified that the physical security plan was being implemented in accordance with the station security plan.

   The NRC inspectors observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection controls. The NRC inspectors walked down the accessible portions of the Unit 1 and 2 emergency feedwater systems, Unit 2 diesel generator fuel oil systems, and

Unit 2 "B" train of Low Pressure Safety Injection to verify operability. The inspector also witnessed portions of the radioactive waste system controls associated with radwaste shipments and barreling.

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

a. Unit 2 Reactor Protection Channel Not Tripped When Required

Unit 2 Technical Specification 3.3.1.1 requires that the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be operable. Linear Power Level - High is functional unit number 2 of Table 3.3-1. This table requires a minimum of 3 operable High Linear Power Level channels in Modes 1 and 2, and it refers to action statement 2; which states, in part:

"With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

b. Within one hour, all functional logic units receiving an input from the inoperable channel are also placed in the same condition (either bypassed or tripped, as applicable) as that required by a. above for the inoperable channel."

On March 20, 1982, at 0250 hours, Channel C Local Power Density-High, and DNBR-Low were bypassed due to an apparent problem with the Channel C Nuclear Instrumentation power supply. Repair efforts during the weekend were not successful, and at 0235 hours, on March 22, 1982, High Linear Power Level, High LPD, and Low DNBR were tripped in Channel C in accordance with action statement 2. Troubleshooting/repair efforts continued during the day on March 22, 1982, and at about 1530 hours, the Nuclear Instrument drawer on Channel C Reactor Protective System was turned off to facilitate efforts to locate a suspected short.

At 1610 hours, the NRC inspector observed during a control room tour that Channel C High Linear Power Level was not tripped and that the drawer was de-energized. The Shift Supervisor was not aware that this channel was not tripped. He checked with an instrument technician and found that the trip signal had been removed when the drawer was turned off. After discussions and consultations, the channel was tripped at 1630 hours by lowering the trip setpoint to near zero.

This is an apparent Technical Specification violation. (368/8205-01)

b. Lifted Leads in the Unit 2 Core Protection Calculator (CPC) Cabinets

The NRC inspector inspected the CPC cabinets in the CPC room on elevation 404 of the Unit 2 Auxiliary Building. In the front and back of the four cabinets, several leads were observed to be lifted without being marked or tagged to indicate the reason for being lifted. Most of these lifted leads had exposed terminals; but several had the terminals clipped off, and others had various types of tape over the terminals. The NRC inspector reviewed drawings and design changes to determine the reason for lifting 30 leads in the "A" and "B" cabinets. These leads had been lifted during the performance of two design changes, IDCR 2-79-95 and DCP 79-2150.

The NRC inspector also observed Jumper and Bypass Tag Number 144, dated January 10, 1978, on leads attached to terminals 4 and 5 of terminal board OTB 3 in the "B" cabinet. Data on this tag could not be located in the Jumper and Bypass Log.

This item will remain open (368/8205-03) pending licensee action indicated below:

- Development and implementation of standard practice for proper handling of leads lifted during performance of a design change

- Application of the standard practice to the lifted leads in the CPC cabinets

- Determination of the continued applicablity of the modification identified by Jumper and Bypass Card Number 144 on OTB 3, terminals 4 and 5 and incorporation into a design change, if necessary.

c. Unit 2 Control Room Operator Awareness

During a routine tour of the Unit 2 Control Room on March 15, 1982, the NRC inspector noted that the three main condenser steam dump valves (2CV-0302, 2CV-0303, and 2CV-0306) were in "manual" mode of operation, as opposed to being in their normal "auto" mode. By being in "manual" mode, these condenser steam dump valves would not always automatically open to provide a heat sink in the event of loss-of-load transients but would, in some cases, require operator action to open.

However, when questioned about the steam dump status by the NRC inspector, the Shift Supervisor and the other Control Room operators were unaware that the steam dump valves were no longer in the "auto" mode. Although these condenser steam dump valves are not considered safety-related, they could have a significant impact on plant response during certain transients, so operator awareness of their status is important. The NRC inspector discussed his concern regarding this lack of operator awareness of an important, but nonsafety-related, plant system status during an exit interview with licensee representatives.

3. <u>Monthly Surveillance Observation</u> (Units 1 and 2)

The NRC inspector observed the Technical Specification required surveillance testing on the electric motor driven emergency feedwater pump, (2106.06 Supplement 2), and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with Technical Specifications and procedure requirements, that test results were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The inspector also witnessed portions of the following test activities:

- Moderator Temperature Coefficient at Power (2302.09)
- Steam Driven Emergency Feedwater Pump Test (1106.06 Supplement II)
- Test of Control Room Emergency Ventilation System (Unit 1 Technical Specification 4.10)
- Control Element Assembly Calculator Number 2 Monthly Test (2304.109)
- Emergency Diesel Generator Monthly Test, Unit 2 (2104.36 Supplement I)
- Response Time Testing of Reactor Coolant RTDs (2304.45)
- Diesel Fire Pump Monthly Test (1104.32 Supplement II)

- Emergency Diesel Generator Monthly Test, Unit 1 (1104.36 Supplement I)

- "B" Low Pressure Safety Injection Pump Monthly Test, Unit 2 (2104.40)

No violations or deviations were identified.

4. Monthly Maintenance Observation (Units 1 and 2)

Station maintenance activities of safety-related systems and components listed below were observed/reviewed 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 operations 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 assure that priority is assigned to safety related equipment maintenance which may affect system performance.

The following maintenance activities were observed/reviewed;

- Reset Unit 1 Anticipatory Reactor Trip Bypass Bistable Setpoints (J.O. 10313)

- Troublesheet/Repair Channel C Excore Nuclear Instrument - Unit 2 (J.O. 25040)

- Modifications to the Waste Gas Collection Header - Unit 1 (DCP 79-1035 and DCP 80-1183)

- Repair of High Pressure Safety Injection Block Valve 2CV-5056-2 (J.O. 25097)

No violations or deviations were identified.

5. I.E. Bulletin 80-06 (Unit 2)

I.E. Bulletin 80-06, entitled "Engineered Safety Feature (ESF) Reset Controls," was issued on 'arch 13, 1980. Action item 2 of this bulletin required the following:

"Verify the actual installed instrumentation and controls at the facility are consistent with the schematics reviewed in Item 1 above by conducting a test to demonstrate that all equipment remains in its emergency mode upon removal of the actuating signal and/or manual resetting of the various isolation or actuation signals. Provide a schedule for the performance of the testing in your response to this bulletin."

The licensee's response to this item for Unit 2 was provided in a letter to the NRC dated June 18, 1980. This letter stated:

"A test has been completed to verify that the actual installed instrumentation and controls are consistent with the schematics reviewed in response to Item 1. In all cases, the schematics and test results agreed as to which components would remain or change upon removal of the ESF signal."

Work Plan 2409.14, "Engineered Safety Features System Actuation Test per I.E. Bulletin 80-06," was prepared to perform the above test. This procedure was partially performed in April 1980. In an exit interview on December 4, 1981, the NRC inspector stated that he was unable to locate a record of the completion of Work Plan 2409.14 and asked if the test had ever been completed. The NRC inspector was informed by a licensee engineer on December 29, 1981, that Work Plan 2409.14 had not been completed. A licensee internal memorandum, (ANO-82-0776), dated February 6, 1982, stated:

"During the last refueling outage on Unit 2, many modifications were completed which made the Work Plan 2409.14 for testing incorrect. The testing done after all modifications were complete was completed by the Integrated ES Test 2304.127 which had been updated to include all modifications."

Procedure 2304.127, referred to above, is entitled "Engineered Safety Features System Response Time Test." This test was performed on July 1, 1981, but it did not determine whether or not, upon the reset of an ESF actuation signal, all associated safety-related equipment remained in its emergency mode.

The NRC inspector concluded that the test required by item 2 of I.E. Bulletin 80-06 has not been performed for ANO-2, contrary to the statement in the licensee's letter of June 18, 1980. This is an apparent deviation from a written licensee commitment to the Commission. (313/8205-02)

The licensee's letters of June 18, 1980, and January 26, 1981, to the NRC stated that containment sump isolation valves, 2CV-5647-1 and 2CV-5648-2 would be modified prior to startup following the 1981 refueling outage. The two valves have not been modified, and the licensee response to I.E. Bulletin 80-06 should be updated. The NRC inspector informed the licensee of the need to correct his bulletin response regarding these two valves in an exit interview on December 4, 1981.

I.E. Bulletin 80-06 was discussed again in exit interviews on March 19 and March 25, 1982. Licensee representatives stated that the incorrect statement in their letter of June 18, 1980, was caused by a misinterpretation of information by personnel in the Corporate Office. Other items discussed at these meetings included:

- Progress in the development of an improved licensee commitment tracking system
- The need for review of corporate correspondence with the NRC by knowledgeable ANO site personnel
- The necessity of a supplemental response to IEB 80-06, providing a commitment and schedule for the testing required by the bulletin, and providing justification for not modifying valves 2CV-5647-1 and 2CV-5648-2.

In response to IEB 80-06, the following Unit 2 valve control circuits were modified:

Design Change	Valve Number	Valve Name
80-2201	2CV-5001-1	Safety Injection Tank Drain Valve
80-2201	2CV-5021-1	Safety Injection Tank Drain Valve
80-2201	2CV-5041-2	Safety Injection Tank Drain Valve
80-2201	2CV-5061-2	Safety Injection Tank Drain Valve
80-2102	2CV-5628-2	Safety Injection Recirculation Valve
80-2102	2CV-5672-1	Containment Spray Pump Recirc.
80-2102	2CV-5673-1	Containment Spray Pump Recirc.
80-2102	2CV-5126-1	HPSI Pump Recirc.
80-2102	2CV-5127-1	HPSI Pump Recirc.
80-2102	2CV-5128-1	HPSI Pump Recirc.
80-2102	2CV-5123-1	LPSI Pump Recirc.
80-2102	2CV-5124-1	LPSI Pump Recirc.
80-2102	2CV-56[illegible]-1	Containment Sump Suction
80-2102	2CV-5650-2	Containment Sump Suction
80-2102	2CV-0714-1	E.F.W. Flush Isolation Valve
80-2102	2CV-0798-1	E.F.W. Flush Isolation Valve

These design changes were performed so that the valves would remain in their emergency positions following reset of an Engineered Safeguards (ES) signal; and operator action would be required to reposition the valves following ES signal reset. The NRC inspector reviewed the above Design Change Packages, their implementing Job Orders, and records of post-modification testing. Post modification testing for these design changes was performed in accordance with Work Plans 2407.29 and 2407.24.

This bulletin remains open for Unit 2 pending resolution of the deviation discussed above and licensee submittal of a supplemental bulletin response as discussed above.

6. I.E. Bulletin 80-06 (Unit 1)

The licensee's response to this bulletin was dated June 18, 1980. This response stated that the drawing review and test required by the bulletin had been completed. The test used to verify that the systems actually responded in a manner consistent with the schematic drawings was Work Plan 1409.10, and was completed on January 24, 1980. The NRC inspector reviewed the records of this test.

In response to IEB 80-06, the following Unit 1 valve control circuits were modified:

Design Change	Valve Number	Valve Name
80-1041	CV-2214	Letdown Cooler Isolation Valve
80-1041	CV-6203	Chilled Water Isolation Valve
80-1055	CV-2630	Main Feedwater Isolation Valve
80-1055	CV-2680	Main Feedwater Isolation Valve
80-1042	CV-2100 thru 2108	Penetration Room Isolation Valves
80-1042	CV-2111 thru 2116	Penetration Room Isolation Valves

These design changes were performed so that the valves would remain in their emergency positions following reset of an Engineered Safeguards (ES) signal; and operator action would be required to reposition the valves following ES signal reset. The NRC inspector reviewed the above Design Change Packages, their implementing Job Orders, and records of post-modification testing.

No violations or deviations were identified; however, this matter will remain open pending review of supplemental information provided relative to Unit 2.

7. Licensee Event Report (LER) 313/82-003/03L-0 (Unit 1)

This LER reported that on January 28, 1982, only one of the three primary makeup pumps (high pressure injection pumps) was operable. This occurred when the C pump was tagged out for changing the oil in the motor bearings, but the oil was changed in the A pump motor bearings. This incident was attributed to inadequate communications and a violation of the Hold and Caution Card Procedure.

This is an apparent violation of Technical Specification 3.3.2 (313/8205-01)

8. Follow Up On Plant Trip

At 2228 hours on March 7, 1982, Unit 2 tripped from 70% reactor power and subsequently experienced a Safety Injection Actuation Signal (SIAS). The transient began with a main turbine runback that resulted from a failure in the main generator cooling system. As a result of the turbine runback, reactor coolant system (RCS) pressure increased to 2340 psig, the steam generator safety valves opened, and the main condenser steam dumps opened; however, the reactor did not trip. At this point, the problem with the main generator cooling system was resolved, and the operators attempted to load the main turbine back down. This sudden increase in turbine load caused steam pressure to drop, which resulted in a rapid increase in steam generator levels ("swell"), and this, in turn, caused the reactor to trip when the high steam generator level setpoint was exceeded. Immediately after the reactor trip, RCS pressure decreased rapidly to the SIAS setpoint due to the continued RCS cooling caused by the open steam generator safety valves and main condenser steam dump valves. Because the steam generator safety valves remained open for only a very short period of time after the reactor trip, and because the operators took prompt action to manually shut the condenser steam dump valves, RCS pressure did not decrease below approximately 1700 psig and then increased rapidly back to above the SIAS reset setpoint. At all times, RCS pressure remained above the shut-off head of the High Pressure Safety Injection pumps.

The NRC inspcctor reviewed all available data obtained during the transient, including computer printouts, recorder traces, log book entries, and the licensee's transient analysis report. Additionally, the NRC inspector interviewed personnel on duty at the time of the event. The NRC inspector verified that all required engineered safety features were operable during the event, all automatic actions taken by plant equipment were correct, manual actions taken by plant personnel were appropriate, operator actions to reset SIAS and to secure low and high pressure safety injection were in accordance with NRC guidelines, and the licensee notified the NRC Operations Center as required by 10 CFR 50.72. The NRC inspector also assessed licensee management's review of the transient and found it to be sufficiently complete.

No violations or deviations were identified.

9. Exit Interview

The NRC inspectors met with Mr. J. P. O'Hanlon (Plant General Manager) and other members of the AP&L staff at the end of various segments of this inspection. At these meetings, the inspectors summarized the scope of the inspection and the findings.