U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
REGION IV

Report Nos. 50-313/78-13
50-368/78-21

Docket Nos. 50-313 License No. DPR-51

50-368 Construction Permit No. CPPR-89/
License No. NPF-6

Licensee: Arkansas Power and Light Company
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: August 1-4 and 8-10, 1978

Inspectors: T. Westerman 8/21/78
T. F. Westerman, Reactor Inspector Date

M. W. Dickerson / for 8/22/78
N. C. Boyter, Reactor Inspector Date

Robert G. Spangler 8/21/78
R. G. Spangler, Reactor Inspector Date

Approved By: M. W. Dickerson / for 8/22/78
G. L. Madsen, Chief, Reactor Operations & Date
Nuclear Support Branch

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Inspection Summary

Inspection on August 1-4 and 8-10, 1978 (Report No. 50-313/78-13)
Areas Inspected: Routine, announced inspection of Technical Specification surveillance calibration, non-Technical Specification surveillance calibration and surveillance of core power distribution limits. The inspection involved 34 inspector-hours on-site by three NRC inspectors.
Results: Within the three areas inspected, no items of noncompliance or deviations were identified.

Inspection on August 1-4 and 8-10, 1978 (Report No. 50-368/78-21)
Areas Inspected: Routine, announced inspection of items preparatory to Operating License issuance and Construction Deficiency reports. The inspection involved 52 inspector-hours on-site by three NRC inspectors.
Results: Within the three areas inspected, no items of noncompliance or deviations were identified.

DETAILS

1. Persons Contacted

Arkansas Power & Light Company Employees

L. Alexander, QC Engineer
B. A. Baker, ANO-2 Operations Supervisor
R. L. Bata, QA Engineer
I. Butler, Electrical Maintenance Supervisor
T. N. Cogburn, Nuclear Engineer
E. C. Ewing, Production Startup Supervisor
D. R. Hamblin, QC Engineer
T. Holcomb, Scheduler
P. Jones, Maintenance Supervisor
J. Lowman, Unit 2 I&C Supervisor
G. H. Miller, Acting ANO Plant Manager
B. Neal, Unit 1 I&C Supervisor
S. Petsel, Licensing Engineer
J. Robertson, ANO-1 Operations Supervisor
P. Rodger, Unit 1 Nuclear Engineer
S. M. Strasner, QC Engineer
B. A. Terwilliger, Supervisor Plant Operations
D. Trimble, Training Coordinator

2. Follow Up on Previously Identified Items (Unit 2)

(Closed) Open Item (Licensee Commitment, Letter from D. H. Williams (AP&L) to J. F. Stolz (NRC), dated April 11, 1978): Requirements for Loss of Off-Site Power Test.

The inspector reviewed procedure 2.800.01, Appendix BB, Loss of Off-Site Power, and determined that the commitments made via the above letter had been implemented.

(Closed) Open Item (NRC Staff Requirement, Summary of September 15, 1977 Outstanding Items Meeting): CESEC Code Verification.

The licensee has approved procedure 2.800.01, Appendix XX to collect the necessary data in order to verify the CESEC results for the three transients specified by the staff at the above meeting.

(Closed) Open Item (Inspection Report 78-15, paragraph 4): Comments on the Emergency Feedwater System/Waterhammer Test.

The licensee has amended procedure 2.650.12 to include the comments noted in the above inspection report.

(Closed) Open Item (Inspection Report 78-05, paragraph 10): Addition of "later entries" to post core test procedures.

The licensee has all "later entries" information in-house (with the exception of the INCA options) and is proceeding to update the required procedures. The INCA options are forthcoming from CE.

(Open) Open Item (Inspection Report 78-12, paragraph 9): During inspection 50-368/78-12, a list of plant procedure discrepancies was identified. In follow up to these items, the inspector found most of these discrepancies to have been resolved either by revision of the affected procedure and by formal notification to NRR (AP&L letter of July 6, 1978 to NRR) that sections of the FSAR will be amended to Amendment 48. Amendment 48 assures that the as-written plant procedure now conforms to the FSAR commitments in the area of Safety Injection Tank Isolation Valve Position and Shutdown Rod withdrawal during plant heatup and cooldown.

The remaining discrepancies in the plant procedures include the following:

- Plant Procedures 2304.37 - 2304.40: Revision of procedures is necessary to include current Technical Specification Plant Protective System Set Points.

- Plant Procedures 2304.92 - 2304.95: Procedure to be revised to include requirements that addressable constants be compared before and after test listing.

- Plant Procedure 2203.12: Procedure is only partially revised.

No items of noncompliance or deviations were identified.

3. Construction Deficiency Reports (Unit 2)

The status of the outstanding construction deficiency reports is as follows:

Qualification Testing of Electrical Penetration Assemblies

This item is under review by NRR.

Potential for Feedwater Pipe Failure Within the Containment Penetration Room

Corrective action is in progress as documented in the AP&L letter of July 12, 1978 and is to be completed prior to Mode 2 operation.

EPG Ball Valve Blocking Lever Failure

The inspector verified that corrective action had been completed as documented in the AP&L letter of July 25, 1978. All action on this item is considered complete.

LPSI and Containment Spray Stop Check Valves

Corrective action is in progress with respect to the failure of the LPSI and Containment Stop Check Valves to fully open. This item is to be completed prior to Mode 4 operation.

4. Technical Specification Calibrations (Unit 1)

The inspector verified for a selected sample of required calibrations that:

a. The Technical Specification for frequency of calibration had been met.

b. Approved and technically adequate procedures were used.

c. Trip set points were as specified in Technical Specifications.

d. Qualified individuals performed the calibrations.

e. Selected test instruments used during calibrations were certified and traceable to an independent testing organization or standard.

The inspection sample consisted of the following procedures:

(1) 1304.41 - Flux/RC Flow Channel Calibration

(2) 1404.01 - Main Steam Safety Valve Set Point Check

(3) 1304.52 - RC Pressure Channel - HPI System

(4) 1304.10 - Core Flood Tank Level and Pressure

(5) 1304.49 - Reactor Building Analog Channel Calibration

(6) 1304.90 - Protective Relaying Interlocks and Circuitry

All procedures reviewed were performed during the 1978 refueling outage. The calibrations appeared to be performed in accordance with approved procedures by competent personnel. The inspector identified no items of noncompliance or deviation in this area.

5. Incore Power Distribution (Unit 1)

The inspector determined that the licensee complied with the Technical Specifications related to the incore power distribution limits, surveillance and calibration requirements. The materials reviewed included:

a. Procedure 1302.15, Core Performance Monitoring and Fuel Management Data Collection.

b. Procedure 1304.48, Core Power Distribution.

c. Operators Quadrant Tilt and Imbalance Log for the month of July.

d. The computer Daily Log for the month of July.

e. Core maps taken on June 30, 1978, July 21, 1978 and July 31, 1978.

f. B&W Topical Report 10123, Nuclear Applications Software Package.

g. Procedure 1304.32, Power Range Linear Amplifier Calibration at Power.

h. Procedure 1302.03, Periodic Incore Detector Calibration.

The reviewed logs indicated that the quadrant power tilt and power imbalance had been maintained within the Technical Specification limits and had been recorded at the required frequency. The power/flow imbalance limits and other pertinent data base parameters specified in B&W 1471, "ANO-1 Cycle 3 Reload Report," had been updated in the plant computer and successfully verified by a B&W furnished test case. Although this was not completed under plant procedural controls, the Nuclear Engineer indicated that further software updates will be made under the recently approved software control procedure 1005.09.

The core maps generated every 10 EFP days as required by the Technical Specifications indicate that the hot channel factors and the linear heat generation rate are within the specifications design basis values.

In reviewing the calibration procedures for the nuclear instrumentation that supports the computer software specified in B&W 10123, the inspector noted the following two items.

(1) The factor 10 E-09 in the formula specified in procedure 1302.03, step 6.1.18, for calculating incore detector leakage correction factors is incorrect and should be 1E-09. This is an open item.

(2) The Nuclear Engineer indicated that it is not necessary to make the flux dependent background correction to the Self Powered Neutron Detectors (SPND) used at ANO-1, as these detectors are constructed such that the net current due to background events occurring in the shield and inner conductor for each SPND cancel.

No further items were identified by the inspector in the above areas.

6. Calibration - Not T/S Specified (Unit 1)

a. Scope

The inspector reviewed the methods used to calibrate components which are either referenced in the Technical Specifications but do not have specific surveillance requirements or components which have safety related functions, as listed below:

(1) Borated Storage Tank Temperature - Calibrated per 1304.12, R2, PC-1, Borated Water Storage Tank Level and Temperature Instructions. Surveillance Test.

(2) High Pressure and Low Pressure Injection Pumps Discharge Flow - Calibrated per 1304.13, R3, PC-1, High and Low Pressure Injection Flow Instructions. Surveillance Test.

(3) Condensate Storage Tank Level - Calibrated per 1304.21, R1, Condensate Storage Tank Level Instructions. Periodic Surveillance Test.

(4) Main Feedwater Flow and Temperature - Calibrated per 1304.65, R0, PC-1, SU and Main FW Flow Instrument.

(5) Reactor Building Spray Pump Discharge Pressure and Flow - Calibrated per 1304.66, R0, Reactor Building Spray Flow.

(6) Diesel Generator Day Tank and Emergency F.O. Tank Levels - Day tank level switches only are calibrated; Emergency Tank level is calibrated per 1303.37, R2, Calibration of Barton Model 227 dP indicator.

(7) Diesel Generator Start Air Receiver Pressure - Calibrated per 1303.39, R4, Pressure Indicating Gauge.

(8) Diesel Fuel Oil Transfer Pump Flow - Flow meter not used; instead use rate of change of tank level per applicable surveillance procedure.

(9) Steam Generator Pressure and Temperature - specific procedure is not used.

(10) Decay Heat Removal System Relief Valve - No procedure available.

b. Findings

(1) Item (9) above, for which there is no specific procedure, is calibrated per the general procedure 1302.01, General Process Instrument Calibration Procedure, R1. This procedure was previously reported to be deficient due to not requiring the use of manufacturers instructions and not requiring that acceptance criteria be specified. The licensee indicated that their procedure would be revised to provide definitive procedure, data sheets, and acceptance criteria. This was identified as unresolved item 7711-2. The procedure has not been revised and item 7711-2 is still unresolved. The inspector was informed that an approved procedure is to be issued by September 14, 1978.

(2) With regard to item (10), the ANO Operational Readiness Program pursuant to ASME, Section XI, is under review by NRR. The testing of safety and relief valves code class 1, 2 and 3 is being included in this review.

7. Exit Meeting

Exit meetings were conducted on August 4 and 10, 1978 with Mr. Miller and other members of the plant staff. The inspectors discussed the scope of the inspection and summarized the inspection findings which are detailed in this report.