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

Report No. 50-313/78-04

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

Licensee: Arkansas Power & LIght Company
Post Office Box 551
Little Rock, Arkansas 72203

Facility Name: Arkansas Nuclear One, Unit 1

Inspection At: Arkansas Nuclear One Site, Russellville, Arkansas

Inspection Conducted: February 13-16, 1978

Inspectors: W D Johnson for 3/8/78
E. H. Johnson, Reactor Inspector Date

W D Johnson 3/8/78
W. D. Johnson, Reactor Inspector Date

Approved By: G L Madsen 3/9/78
G. L. Madsen, Chief, Reactor Operations and Nuclear Support Branch Date

Inspection Summary

Inspection on February 13-16, 1978 (Report No. 50-313/78-04)

Areas Inspected: Routine announced inspection of refueling activities; follow-up on reactor coolant pump seal instrument line leak; follow-up on hydraulic snubber failures; tour of accessible plant areas; fuel handling area ventilation system operability; and containment isolation valve operability. The inspection involved fifty-eight inspector-hours on site by two inspectors.

Results: Of the six areas inspected no items of noncompliance or deviations were noted in three areas. Three items of noncompliance (infraction - failure to follow procedure in that source range count rate was not recorded following fuel element moves as required by the licensee's refueling

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procedure - DETAILS, paragraph 3; infraction - failure to perform safety related activities under suitably controlled conditions in that maintainance was permitted on one channel of source range instrumentation while refueling operations were in progress - DETAILS, paragraph 3; infraction - failure to conduct routine functional testing on a containment isolation valve as required by T. S. 4.4.1.4 - DETAILS, paragraph 6; infraction - failure to maintain surveillance procedure as required by T. S. 6.8.1 - DETAILS, paragraph 7) were noted in three areas.

## DETAILS

1. Persons Contacted

*L. Alexander, QC Engineer
P. Almond, Reactor Engineer
J. Anderson, Plant Superintendent
*T. Cogburn, Nuclear Engineer
G. DuPriest, Shift Supervisor
*R. Elder, I&C Supervisor
T. Green, Training Coordinator
C. Halbert, Technical Support Supervisor
*D. Hamblen, QC Inspector
J. Maxwell, Shift Supervisor
*J. McWilliams, Planning and Scheduling
H. Miller, Assistant Plant Superintendent
W. Moon, Shift Supervisor
P. Jones, Maintenance Supervisor
J. Lohman, I&C Supervisor
*J. Robertson, Assistant Operations Supervisor
P. Rogers, Reactor Engineer
*S. Strasner, QC Inspector

*Denotes those attending the exit interview.

The inspectors also talked with and interviewed several other licensee employees during the inspection. These included reactor operators, maintenance technicians, and office personnel.

2. Follow-up on Unresolved Item

(Closed) Unresolved Item (UI 7725-33): Certificates of Compliance for cycle 3 fuel.

During inspection 77-25, the inspector reviewed the licensee's preparation for refueling, but the manufacturer's certificates of compliance for the new fuel assemblies could not be located at that time. During this inspection, the inspector reviewed the certificates for the 56 new fuel assemblies and compared their serial numbers with the tags on the fuel status tag board in the control room. No items of noncompliance or deviations were identified.

3. Refueling Activities

a. Preliminary Activities

The inspectors made direct observations and reviewed records as necessary to verify compliance with the surveillance testing, procedural and Technical Specification requirements associated with the following pre-refueling activities:

(1) Refueling machine pre-operational tests

(2) Spent fuel handling machine pre-operational tests

(3) Fuel storage area ventilation system operability tests

(4) Crane testing

(5) Refueling deck radiation monitors operability tests

(6) Communications systems checkout

(7) Stored fuel cooling system operability checks

(8) Reactor building purge isolation system

(9) Reactor building evacuation alarm testing

(10) Automatic containment isolation valves operability tests.

b. Fuel Handling Activities

The inspectors made direct observations and reviewed records as necessary to verify existence of the following conditions and to verify that the activities were conducted as required by Technical Specifications and approved procedures:

(1) Containment integrity was properly established in accordance with Technical Specification 3.8.6 and 3.8.7.

(2) The required containment atmosphere and plant ventilation direct radiation monitors were operable as required by Technical Specification 3.8.10.

(3) Radiation levels in the containment and spent fuel storage areas were monitored as required by Technical Specification 3.8.1.

(4) Neutron flux was monitored as required by Technical Specification 3.8.2 and Operating Procedure 1502.04.

(5) One decay heat removal pump and cooler and its cooling water supply were operable as required by Technical Specification 3.8.3.

(6) The reactor coolant system and the spent fuel pool were sampled at the required frequency and boron concentration was found to be in compliance with Technical Specification 3.8.4.

(7) Communications between the control room and the refueling machine were maintained as required by Technical Specification 3.8.5.

(8) The fuel handling area ventilation system was in operation as required by Technical Specification 3.15.1.

(9) Refueling pool water level was maintained as required by Operating Procedure 1502.04.

(10) Fuel assembly insertion and removal from the core was performed in accordance with Operating Procedure 1502.04.

(11) Control room activities associated with fuel assembly insertion and removal were performed in accordance with Operating Procedure 1502.04.

(12) Core internals, the reactor vessel head, studs, nuts, and washers were stored properly.

(13) Fuel accountability methods were in accordance with Operating Procedures 1502.04 and 1502.05.

(14) Housekeeping on the refueling deck was satisfactory.

(15) The makeup and license requirements of the refueling crew on the refueling deck and in the control room were in accordance with Technical Specification Table 6.2-1.

c. Findings

(1) Testing overload and low load trips on fuel handling machines. These trip functions were tested in accordance with attachments to plant procedure 1502.03, "Preparation for Refueling." These give approximate load values at which the overload and low load trips on the main bridge, the auxiliary bridge, and the spent fuel bridge hoists

should occur and provide for recording the actual load when trip occurs. The procedure contains no acceptance criteria, and the licensee could provide no basis for the trip values given in the procedure.

The licensee has letters from their fuel supplier recommending trip setpoints for overload and low load, but could not determine if the setpoints had ever been adjusted in accordance with the recommendations.

These hoists are not treated as safety related by the licensee, so no records of trip setpoint adjustments have been maintained. The procedural setpoints and the recorded "as found" values are given below.

| | Overload Setpoints | | Low Load Setpoints | |
|---|---|---|---|---|
| | Procedure | As found | Procedure | As found |
| Main Bridge | 2480 lbs. | 2620 lbs. | 2100 lbs. | 1700 lbs. |
| Aux. Bridge | 2480 lbs. | 2650 lbs. | None given | Not recorded |
| Spent Fuel Bridge | 2480 lbs. | 2700 lbs. | 2100 lbs. | 1700 lbs. |

This item remains unresolved pending further review by the licensee and the inspector. (Unresolved Item UI 7804-01)

(2) Recording source range count rate. While observing refueling operations from the control room, the inspector observed that the licensee was not maintaining 1/M plots or otherwise recording the neutron flux level after insertion of a fuel assembly into the core. Plant Procedure 1502.04 requires that neutron count rate be recorded in the chronological Log following loading of each fuel assembly into the core and prior to subsequent loading. The operator stated that his practice was to take a ten-second count on the counter scaler after loading each fuel assembly and to look for any unusual increase. The licensee's failure to implement and maintain compliance with a procedure designated in Section B.10 of Regulatory Guide 1.33 is contrary to Section 6.8.1 of the Technical Specifications.

(3) Maintaining control of activities during refueling. The inspectors were in the control room observing refueling operations and were observing the indications of the

source range nuclear instruments when the source range channel A instrument was de-energized. Investigation revealed that the channel was de-energized by technicians performing an annual test of RPS channel A in accordance with plant procedure 1304.41 and maintenance work order 8451-R2. This caused a loss of indication on the counter scaler which was not immediately understood by the refueling operator in communication with the reactor building.

The inspector pointed out to the plant operator that Technical Specification 3.8.2 requires that core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors when core geometry is being changed. With the channel A source range instrument secured, only one channel remained in operation. After trying to restore counter scaler operation by resetting it, the refueling operator requested that fuel handling operations in the reactor building be suspended. On the plant operator's request, the instrument technicians re-energized source range channel A and bypassed that portion of their procedure.

The inspectors expressed concern to the Shift Supervisor, Assistant Operations Supervisor, and the Plant Superintendant that these two separate activities, refueling and testing of the reactor protective system, were occurring simultaneously without any overall control or coordination. The special precautions section of maintenance work order 8451-R2 for testing the RPS included only "Normal Safety Precautions." The Shift Supervisor had apparently approved this work order without realizing that it could impact on the ongoing fuel handling operations. Subsequent checks by the licensee indicated that during the period of about two minutes while only one source range monitor was in operation, core geometry was not actually being changed. Thus it appears that Technical Specification 3.8.2 was not violated. However, due to the lack of control and coordination of activities, the licensee had been placed in a position in which the violation of this Technical Specification could easily have inadvertently occurred.

Criterion II of Appendix B to 10 CFR 50 requires that activities affecting quality shall be accomplished under suitably controlled conditions, including assurance that all prerequisites for the activity have been satisfied. This is amplified by Section 2.4.2 of the licensee's NRC approved Quality Assurance Manual-Operations. The licensee's failure to control and coordinate activities during refueling operations is an apparent item of noncompliance with the above requirements.

(4) Fuel assembly serial number verification. Plant procedure 1502.04 requires that fuel assembly serial numbers be verified visually prior to insertion into the core. It states that this verification should be performed in the reactor building upender, but may be performed in the spent fuel pool.

The inspector observed that fuel assembly serial numbers were not being verified during the fuel shuffle, either in the reactor building or in the spent fuel pool. A licensee representative informed the inspector that the verification requirement had been met by the performance of an inventory of the new fuel assemblies in the spent fuel pool prior to the fuel shuffle. He added that the core configuration would be independently verified after the completion of the fuel shuffle and that a videotape record would be retained. The inspector had no further questions on this item.

(5) Changes to Refueling Shuffle procedure. The inspectors observed that plant procedure 1502.04, "Refueling Shuffle," provided that changes could be made to the refueling shuffle sequence with the approval of the Senior Reactor Operator in charge of refueling. However, the procedure does not specify that another member of the plant staff must approve the change of provide for the documentation of the change and its approval in accordance with the requirements of Technical Specification 6.8.3.

The inspectors indicated to the licensee representatives at the exit interview that any temporary procedure changes must be accomplished in conformance with the requirements of Technical Specification 6.8.3.

4. Plant Tour

The inspectors toured accessible areas of the plant including control room, cable spreading room, battery rooms, diesel generator rooms, auxiliary building and reactor building. The inspectors noted numerous valve leaks in the auxiliary building and reactor building. These leaks were evidenced by boric acid crystal buildup at the valve stem or body to bonnet gasket area. A licensee representative indicated that such leaks were noted on maintenance lists for correction during the refueling outage. One rigid hanger in the auxiliary building was noted to be loose from its pipe. A licensee representative accompanying the inspectors noted the location of this hanger for correction.

The inspectors noted that numerous channels on the boric acid system heat tracing alarm panel were alarming. For channel 2722 both redundant channels were in alarm. A review of drawing M288 and associated heat tracing isometrics revealed that this section was on the boric acid mixing tank level transmitter, a non safety related portion of piping. The inspector expressed his concern, however, about the numerous channels that were in alarm.

General housekeeping conditions in the auxiliary building and reactor building were noted. The general condition of the reactor building in view of the maintenance and operations in progress appeared to be satisfactory. A licensee representative indicated that following the outage a general cleanup of the reactor building would be conducted.

General housekeeping conditions in several areas of the auxiliary building and spent fuel pool area were noted to be in need of attention. The inspectors indicated that this item will receive further follow-up during subsequent inspections.

No items of noncompliance or deviations were noted in this area of the inspection.

5. Fuel Handling Area Ventilation System

Technical Specification 3.15 requires that the fuel handling area ventilation system be operable whenever irradiated fuel is handled in the spent fuel pool. The inspector reviewed plant records to verify that testing had been performed to assure conformance to technical specification surveillance requirements. Job Order 4301-77-3 was performed on December 14, 1977, to prescribe the testing on the fuel handling area ventilation system. The following results were noted.

| | Test Results | Acceptance |
|---|---|---|
| Methyl Iodide Absorption | 95.62% | >90% |
| Flow Test | 36840cfm | 39000 ± 10%cfm |
| DOP Test | 99.99% | >99% |
| Hydrogenated Halide | 99.99% | >99% |

The inspector was satisfied that the surveillance requirements for the system had been met.

Technical Specification 3.15.1.e. requires that "air distribution shall be uniform within $\pm$ 20% across HEPA filters and charcoal absorbers when tested initially and often after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system . . . ." The inspector requested that records of maintenance on this system be made available so he could determine if this specification had been properly implemented. The inspector was informed that this system was not Q-listed (FSAR Table 1-2) and thus maintenance performed on the system would not be required to be documented by completion of a job order form.

A review of the safety analysis section of the FSAR revealed that on a fuel handling accident (FSAR section 14.2.2.3) it is assured that all gases released to the spent fuel pool area are processed by the fuel handling area ventilation system (HEPA filters and charcoal beds) before release to the environment. The analysis further indicates that 90% of all gases released to the spent fuel pool area are removed by this ventilation system.

FSAR section 1.4.1 specifies that "Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified then designed, fabricated and erected to quality standards that reflect the importance of the safety function to be performed. . . ."

The inspector discussed the intent of the above FSAR commitment with licensee representatives at the exit interview. The inspector noted that the testing performed on the system is handled as a Q-list activity and that replacement parts are procured in a manner similar to the control room and reactor building ventilation systems, both of which are Q-listed systems, and are very similar to the fuel handling area ventilation system. The inspector agreed that an attempt to backtrack procurement and construction records for the system in an attempt to upgrade the system to Q-List level would not be feasible nor necessary. However the inspector indicated that this system should be added to the Q-List to satisfy the FSAR commitment indicated above so that all future maintenance activities on the system are properly controlled. The inspector indicated that this item would remain unresolved pending the licensee's review of this area. (Unresolved Item UI 7804-02).

No items of noncompliance or deviations were identified at this time in this area of the inspection.

6. Containment Isolation Valves

Technical Specification 3.8.7 requires that during refueling, all containment isolation valves shall be operable or at least one valve shall be closed. Technical Specification 4.4.1.4 requires that all containment isolation valves be stroked to their isolation position once each three months unless such action is not practical during plant operation, in which case they shall be tested every 18 months. The licensee performs this testing under two surveillance procedures 1304.70, "Reactor Building Isolation Valve Stroke Test," for all containment isolation valves that are capable of being stroked quarterly and 1304.08, "Integrated ES Logic Test," for all containment and engineered safeguards valves. The inspector reviewed these procedures in detail and noted that valve CV-1065 Quench tank condensate isolation valve is not included for quarterly stroking in procedure 1304.70. Procedure 1103.05, "Pressurizer Operation," indicates that this valve is normally open while its isolation position is closed. Discussions with plant operations personnel indicated that there is no operational difficulty associated with the routine functional test of this valve.

This failure to stroke valve CV 1065 to its isolation position each three months is in noncompliance with Technical Specification 4.4.1.4. indicated above.

No other items of noncompliance or deviations were noted in this area of the inspection.

7. Leak from Reactor Coolant Pump Seal Pressure Transmitter Instrument Line

During the shutdown and cooldown on February 3, 1978, to commence the current refueling outage, a reactor compartment entry was made to determine the status of systems and components in the reactor compartment while at normal operating pressure. During this entry a leak was noted on the C reactor coolant pump number 2 seal pressure transmitter instrument line. This leak had been present for some period of time and led to high levels of reactor building surface contamination and some airborne contamination. This occurrence was investigated and reported on an investigation report 50-313/78-05.

During this inspection, the inspector reviewed plant records concerning the leak to verify the licensee's calculations regarding leak rate and the impact on the plant. The following records were reviewed:

Procedure 1103.13, "Determination of Reactor Coolant Leakage," November 2, 1977, thru November 22, 1977, and December 8, 1977, thru February 3, 1978.

Computer NSSS periodic log December 8, 1977, thru February 3, 1978, (inspector independently calculated leak rates from computer information and compared this to licensee calculations on procedure 1103.13)
Reactor building sump reading December 8, 1977, thru February 3, 1978, (second independent means of estimating reactor coolant leak rate)
Reactor building gaseous activity monitor readings December 8, 1977, thru February 3, 1978.

Based on a review of this data, the inspector determined that the instrument line leakage was approximately 0.5 gallons per minute above the background losses of approximately 0.45 gpm. This leakage is within the limits of Technical Specification 3.1.6 (1 gpm unidentified leakage).

During a review of the licensee's calculated leak rates as determined by procedure 1103.13, the inspector noted that on numerous occasions the "Leak Rate" calculated in block 7 of the date sheet exceeded 1 gpm. This calculated leakage rate was determined from the change in makeup tank level in gallons divided by the time interval of the readings and correcting this for pressurizer level change, changes in average coolant temperature and reactor power. The total leakage arrived at by following this primary mass balance includes both unidentified and controlled leakage from the primary. Controlled leakage consists of leak sources that are known, collected and measured. Such sources are reactor coolant pump seal leakage (measurable at the seal) which is directed to the quench tank, and pressurizer relief/safety valve seat leakage which is directed to the quench tank. On those occasions where the block 7 leak rate exceeded the 1 gpm limit for unidentified leakage, the licensee subtracted the controlled leakage leak rate from the total and was able to demonstrate that total unidentified leakage remained below the T.S. limit.

However, procedure 1103.13 does not address the calculation of controlled leakage. The procedure does not specify that quench tank or reactor coolant pump seal leak detector readings be recorded at the beginning and end of the data period, and the correction factors are not specified. For the correction to the leak rate that is captured by the quench tank no precautions are specified as to the appropriate quench tank conditions that must be maintained. For instance, no quench tank cooling by condensate to be initiated during the measurement period, or such other factors that would create a nonconservative measurement of the controlled leakage being captured by the quench tank.

As a result of this the inspector noted that on all but a few leak rate data sheets the controlled leakage valve was recorded without recording appropriate raw data (quench tank level readings before and after, etc.). The inspector was informed that verbal instructions had been given to the operators performing the leak rate calculation on how to handle the controlled leakage however, the procedure was not changed to reflect this. For one daily leak rate calculation, the inspector was able to determine that the controlled leakage value being subtracted from the block 7 "leak Rate" was the sum of quench tank level change and reactor building sump level change. This was apparently done since reactor coolant pump seal leakage that was being experienced at this time was being directed to the sump. This is erroneous since the reactor building sump is an open system collecting unidentified leakage in addition to known sources (controlled leakage).

Technical Specification 6.8.1 requires that procedures be implemented and maintained to cover surveillance and testing of safety related systems. Procedure 1103.13 is such a procedure. The failure to maintain procedure 1103.13 to specify the method of properly determining controlled leakage is in noncompliance with this requirement.

No other items of noncompliance or deviations were noted in this area of the inspection. The following additional items were noted.

Technical Specification 3.1.6.7 requires three operable systems for leak detection utilizing different detection methods. One such system shall sense radioactivity and shall consist of a gaseous detector and a particulate detector. The technical specifications allow for both of these channels to be inoperable if grab samples are taken every seventy-two hours. The on line particulate detector has been inoperable for approximately three years. The licensee did not indicate to the inspector any intention of restoring this channel operability. A licensee representative indicated that periodic grab samples were being taken. The inspector indicated that it was clearly the intent of the specification that such on line monitoring capability be employed. This item will be designated as unresolved. (Unresolved Item UI 7804-03).

The inspector reviewed the shift supervisor's logbook to determine what log entries had been made concerning the leak discovered on February 3, 1978. The inspector could find no indication in the log either for the shift that discovered the leak or for the next shift when the leak was isolated. This item was discussed with a licensee representative who agreed with the inspector's concerns that the shift supervisors log should contain entries regarding such findings as a means communicating plant status to subsequent shifts and to station management.

The licensee representative indicated that some improvement in the completeness of log entries was desirable. This item will remain open for further review during a future inspection.

8. Hydraulic Shock Suppressor (Snubber) Testing

The inspectors reviewed the snubber functional testing program that was being conducted by the licensee. A licensee representative reported that several of the snubbers had failed to meet the test acceptance criteria for lockup velocity or bleed rate. He stated that AP&L would remove, test, adjust/repair as necessary, and re-install all snubbers listed in Technical Specification Table 3.16-1. He further stated that a Licensee Event Report would be submitted at the completion of the testing program to report all snubber failures noted.

The inspector reviewed six as-built hanger drawings to verify that applicable snubbers were included in the Technical Specification Table. He found no discrepancies, but was informed that the licensee had identified two additional snubbers that should be listed. The licensee representative stated that a facility license change would be proposed which would add these snubbers to Table 3.16-1. He also stated that AP&L would review the difficulty of removing those snubbers listed in Table 3.16-1 as "Snubbers Especially Difficult to Remove" and propose changes to the table if appropriate.

No items of noncompliance or deviations were identified.

9. Unresolved Items

Unresolved items are matters about which more information or actions by the licensee are necessary to ascertain whether they are acceptable items, items of noncompliance, or deviations. These items are identified in paragraphs 3, 5 and 7 of this report.

10. Exit Interview

The inspector met with licensee representatives (denoted in paragraph 1) at the conclusion of the inspection on February 16, 1978. The inspectors summarized the scope and findings of the inspection.