U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-409/86002(DRP)

Docket No. 50-409 License No. DPR-45

Licensee: Dairyland Power Cooperative
2615 East Avenue - South
La Crosse, WI 54601

Facility Name: La Crosse Boiling Water Reactor

Inspection At: La Crosse Site, Genoa, WI

Inspection Conducted: February 17, 1986 through April 4, 1986

Inspector: I. Villalva

Approved By: D. C. Boyd, Chief
Reactor Projects Section 2D

4/10/86
Date

Inspection Summary

Inspection from February 17, 1986 through April 4, 1986 (Report No. 50-409/86002(DRP))

Areas Inspected: Routine, unannounced inspection by the resident inspector of Licensee Actions on Previous Inspection Findings; Operational Safety Verification; Maintenance Activities; Licensee Event Reports Followup; Plant Trips; Refueling Activities; and Onsite Review Committee. The inspection involved a total of 102 inspector-hours onsite by the NRC resident inspector including a total of 12 inspector-hours during back shifts.
Results: No violations of NRC requirements were noted.

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DETAILS

1. Persons Contacted

*J. Parkyn, Plant Superintendent
H. Towsley, Technical Support Engineer
*G. Boyd, Operations Supervisor
*L. Kelley, Assistant to Operations Supervisor
*L. Nelson, Health and Safety Supervisor
R. Wery, Quality Assurance Supervisor
S. Raffety, Reactor Engineer
P. Bronk, Nuclear Engineer
L. Goodman, Operations Engineer
R. Brimer, Electrical Engineer
D. Rybarik, Mechanical Engineer

The inspector also interviewed other licensee personnel during the course of the inspection.

*Denotes those attending exit interviews during the inspection period.

2. License Action on Previous Inspection Findings

a. (Open) Open Item (409/84009-17) (IPSAR Reference 4.21.5): Develop Procedures to Specify Under Which Conditions Remote Manual Valves Will be Closed. By letter of February 8, 1985 (LAC-10567) to the Office of Nuclear Reactor Regulation, the licensee proposed to install two check valves in series in the sampling line to the shell side of the shutdown condenser. The inspector discussed this with the licensee and noted that a draft "Facility Change" had been prepared in anticipation of receiving approval to install the valves.

   By letter dated March 10, 1986, (John A Zwolinski to James W. Taylor), NRR enclosed a safety evaluation on this item stating that the staff had reviewed the licensee's proposal and had concluded that the planned modification will provide acceptable isolation capability for the shutdown condenser sample line.

   This item, however, remains open pending a final "Facility Change" and verification of installation of the valves.

3. Operational Safety Verification

The inspector observed control room operations, reviewed applicable logs and conducted discussions with control room operators during the period of this report. The inspector verified the operability of selected emergency systems, reviewed tagout records and verified proper return to service of affected components. Tours of the crib house, reactor building and turbine building were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and to verify that maintenance requests had been initiated for equipment in need of maintenance. The inspector by observation and direct interview, verified that the physical security plan was being implemented in accordance with the station security plan.

The inspector observed plant housekeeping cleanliness conditions and verified implementation of radiation protection controls. The inspector walked down the accessible portions of the alternate core spray and high pressure service water systems to verify operability. 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.

4. Maintenance Activities

On March 7, 1986, station service power was being transferred from the normal source to the reserve source as a normal step in the shutdown process then underway for a refueling outage. During the transfer process, Reserve Feed Breaker 1B failed to close, but since Main Feed Breaker 1B opened, a loss of voltage to 2400 V Bus 1B resulted. The loss of 2400 V Bus 1B, in turn, caused the reactor to scram and Emergency Diesel Generator 1B to start and supply the 1B Essential Bus. In addition, Control Rod No. 20 only inserted to about 30% of its full travel during the scram. All other control rods fully inserted to shutdown the reactor.

The immediate maintenance actions taken for the failed breaker included the installation of a spare breaker as the 1B Reserve Feed Breaker. Subsequent corrective actions included a detailed visual examination of the failed breaker, adjusting the contact strokes on Phases A and B of the breaker to increase blade penetration, replacing the breaker's trip latch, and operating the breaker successfully 35 times after these actions were taken.

Failure of Control Rod No. 20 to fully insert was initially diagnosed by the licensee as being due to a faulty clutch. Subsequent troubleshooting revealed that the failure was due to a malfunction of the hydraulic scram motor and that further troubleshooting of the scram motor was beyond the licensee's capability. Consequently, the scram motor was sent to Vickers, the manufacturer, to determine the actual cause of the failure. The paragraph that follows has been extracted from Vickers' report to the licensee describing the failure mechanism:

> "It was noted that the retaining ring in the cylinder block was broken, releasing the pin and knuckle subassembly, one section of the top flange of the cylinder block was broken off, the outer edge of the cylinder block was badly damaged, and the valve block was scored. Examination of the broken parts returned with the unit revealed a broken section of a steel piston (completely foreign to this unit), was among them. Careful inspection of all the parts, exposed an impression of the broken piston in the wall of the angle housing. The loose part had been inside the unit from its original

assembly and had floated inside, undetected at performance test and through many hours of service. However, at one point of operation the fragment shifted into the position which caused the catastrophic failure. How the piston fragment found its way inside the motor is an unexplained happenstance. The motor will be rebuilt under full warranty and will be returned as soon as possible."

Based on the manufacturer's findings and the operational history of identical scram motors used at LACBWR, the licensee has concluded that this event was a random failure having no generic implications for the other scram motors used at LACBWR. The inspector concurs in this finding and considers that the maintenance actions taken for this event are acceptable.

5. Licensee Event Reports Followup

Through direct observations, 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 had been accomplished in accordance with technical specifications.

a. (Closed) LER 86-03: Reactor Scram - Fuse Blown While Adjusting Seal Inject Flow to Control Rod Drive 2. On January 12, 1986, while the reactor was in Operating Condition 2 (startup and heatup) the effluent temperature from Control Rod Drive (CRD) No. 2 was hotter than from the other CRDs. Maintenance had been recently performed on CRD No. 2, and upon investigation, it was determined that a valve on the seal inject supply line to CRD No. 2 had been left closed. At 1107, while the closed seal supply valve at CRD No. 2 was being opened, the seal inject tubing was bumped into and shorted the terminal block mounted on the side of the CRD, causing fuses 29/2 and 55/2 to blow. The blown fuses, in turn, caused the reactor to scram, the high pressure core spray (HPCS) pumps to start, containment building to isolate and the emergency diesel generators (EDG) to start.

   Although the actions taken by the licensee for this specific event warranted its closing, it was held as an open item in a previous report (50-409/85022-02) pending the licensee's evaluation of the generic aspects of the event, (e.g., the plants susceptibility to partial scrams because of the exposed terminals in the immediate vicinity of the CRDs). The licensee has completed its evaluation of this event, and has determined that attempts to insulate or cover the exposed terminals are not justified and that such actions would not only increase personnel exposure but may also be counter productive. The licensee's actions are deemed acceptable and this item is considered closed.

b. (Closed) LER 86-05: Removal of 1A High Pressure Service Diesel from Service for Hose Replacement. At 0840 on January 24, 1986, the diesel-driven 1A High Pressure Service Water (HPSW) pump, which also serves as an Alternate Core Spray (ACS) pump, was removed from service to replace a weeping cooling water hose on the diesel. This conscious and conservative action was taken although removing either one of the two ACS pumps from services causes entry into Section 3.0.3 of the LACBWR Technical Specifications which requires the reactor to be placed in hot shutdown within 12 hours and cold shutdown within an additional 30 hours unless corrective measures are completed first. Since the 1A HPSW/ACS diesel-driven pump was returned to operable status at 1347, well within the 12 hour time limit, reactor shutdown was not initiated.

The actions taken by the licensee regarding this event were completely acceptable and justified; therefore, this event is closed.

6. <u>Plant Trips</u>

The plant trip discussed in Paragraph 4 of this report merely advanced the start time for the refueling outage such that the licensee took no immediate actions for returning to power. Consequently, this report does not address the actions taken by the licensee in returning to power; however, such actions will be covered in the next report after an LER on the event has been issued. In the interim, it can be said that the maintenance actions taken by the licensee for both the breaker that failed to transfer and the subsequent failure of Control Rod No. 20 to fully insert were acceptable.

This event is highlighted because of concerns regarding the potential generic aspects of the control rod drive failure. However, based on the results of Vickers' examination of the failed scram motor, as described in Paragraph 4 and previous operating experience with similar hydraulic motors, this event does not appear to have generic implications.

7. <u>Refueling Activities</u>

Refueling activities were initiated subsequent to the March 7, 1986, reactor trip event discussed above. The inspector observed two shifts of the fuel handling operations (removal and installation) and verified the activities were performed in accordance with the technical specifications and approved procedures; verified that containment integrity was maintained as required by technical specifications; verified that good housekeeping was maintained on the refueling area; and verified that staffing during refueling was in accordance with technical specifications and approved procedures.

As fuel elements were removed from and installed in the core the licensee used a tensiometer, as required by procedures, to assist in determining that the elements were "free" and were not obstructed in any manner. Tensiometer readings are calibrated to the free hanging weight of the elements, and there was a copy of a graph relating readings to weight. The inspector observed refueling personnel using the graph to verify the weights. After the element was moved to the fuel element storage well the inspector observed that fuel element serial numbers were verified and that this information was communicated to the control room. As elements were moved, the Accountability Officer verified each step on a checklist and also communicated this information to the control room. As required by procedure, certain elements were scanned by video camera for damage or crud, and video tapes were made of this scan.

The inspector observed that precautions were taken to prevent foreign objects from falling into the open reactor vessel. A control zone was marked on the floor with tape and anything that was taken into that zone had to be logged by object name and the time. It was then logged out when it was removed from the area. Objects which were needed routinely for refueling purposes (such as the tensiometer) were permanently tied to lanyards so they could not be lost.

On March 15, while control rod handling was in progress, the control rod in position 19 appeared not to be latched to its drive mechanism. Therefore, it was conservatively assumed that the rod had been unlatched since September 1984, when its upper control rod drive mechanism was last installed. All but 2 of the other control rods were subsequently pull tested and found to be latched. The remaining 2 control rods were not readily accessible; therefore, they were not pull tested. In lieu of a pull test, the 2 remaining control rods were determined to be latched by using go-no-go measurements.

Upon reviewing the event, it has been determined that the procedure for latching control rods needs improvement and the tolerance specified on the data sheet is too great. Accordingly, the procedure and data sheet are being revised, and this event is being held as an open item (50/409-86002-01), pending the revision of the latching procedure and associated data sheet.

This event is highlighted because of the concerns regarding the ability of an unlatched rod to scram as well as the nuclear effects of an unlatched rod not being in its programmed position during power operation. These concerns have been alleviated by the results of visual examinations of Control Rod No. 19 and rod following checks conducted during reactor startup. For example, a visual inspection of Control Rod No. 19 revealed clean surfaces on the lower taper of the end stud. These clean surfaces suggest that although the control rod had not been properly seated (latched) to its upper control rod drive mechanism, the latch balls were in contact with the control rod's end stud. The wear pattern further suggests that the upper control drive mechanism's push rod probably pushed the control rod up (inserted) during the past scrams. In addition, the clearance check for the unlatched control rod indicated that it should have been able to follow the push rod whenever it was withdrawn. Further, as part of each reactor startup, a rod following check is conducted after criticality is achieved (i.e., each control rod is inserted until a definite decrease in power level and a negative period is observed, and then the rod is withdrawn to its former position and an increase in power is noticed). The most recent such rod following check was conducted on January 28, 1986.

Based on the above, it is reasonable to assume that Control Rod No. 19 has been following its drive, at least at the critical heights as determined during the rod following checks.

In addition to the usual activities associated with a refueling outage, the licensee also replaced 9 of 29 control rods with new ASEA designed control rods during the current refueling outage. The ASEA designed control rods consist of a cruciform absorber section containing hafnium in the upper tips and boron carbide in the remaining sections of the cruciform. The 9 new control rods were placed in controlling positions Nos. 1 and 6 through 13, inclusive. Position No. 1 is in the center of the core and positions 6 through 13 are located in the second ring group from the center of the core. The remaining 20 old control rods are to be replaced with new ASEA rods, 11 of which are at the site, during future refueling outages. In the meantime, the 9 old control rods and another control rod that was previously removed from the core are being stored in the spent fuel storage pool. The final disposition of these 10 control rods has not yet been determined.

8. Onsite Review Committee

The inspector attended onsite review meetings conducted during the period of this report and examined onsite review functions to verify conformance with technical specifications and other regulatory requirements. These review meetings discussed inspection in the charter and/or administrative procedure governing proposed technical specification changes, violations and corrective actions, proposed facility and procedure changes and proposed tests and experiments conducted per 10 CFR 50.59, and others required by technical specifications.

9. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. A new open item is described in Paragraph 7.

10. Exit Interview

The inspector met with licensee representatives (denoted in paragraph 1) throughout the inspection period and at the conclusion of the inspection and summarized the scope and findings of the inspection activities. The licensee representative acknowledged the findings as reported herein and did not identify such documents or processes as proprietary.