U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-409/81-23(DPI .

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

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

Facility Name: LaCrosse Boiling Water Reactor

Inspection At: LaCrosse Site, Genoa, WI

Inspection Conducted: December 22, 1981 through March 31, 1982

Inspectors: J. F. Streeter for M. W. Branch 7/26/82

J. F. Streeter for W. L. Forney 7/26/82
(December 29, 1981 through January 4, 1982, only)

J. F. Streeter 7/26/82
J. F. Streeter
(February 26, 1982, only)

Approved By: J. F. Streeter
J. F. Streeter, Chief 7/26/82
Projects Branch 2

Inspection Summary

Inspection on December 22, 1981 through March 31, 1982 (Report No. 50-409/81-23(DPRP))

Areas Inspected: Routine resident inspection of Operational Safety; Maintenance Activities; Surveillance Testing; Bulletin Followup; Licensee Event Report Followup; Followup on Plant Scrams; Followup on Open Inspection Items; TMI Task Action Plan Followup; Training and Requalification Training; and Type B and C Leak Rate Testing. The inspection involved a total of 237 inspector-hours onsite by two NRC inspectors including 28 inspector-hours onsite during off-shifts.

Results: Of the ten areas inspected, no items of noncompliance or deviations were noted in eight of the areas. Three items of noncompliance were noted in the other two areas (failure to follow procedures that resulted in unintentional cooldown of the No. 1B Forced Circulation Loop to below Technical Specification limits - Paragraph 7; failure to conduct a 50.59 review and failure to follow procedure during the March 16, 1981, event of power operation on the Main Steam Bypass Valve - Paragraph 8).

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DETAILS

1. Persons Contacted

   *R. Shimshak, Plant Superintendent
   *J. Parkyn, Assistant Plant Superintendent
   *G. Boyd, Operations Engineer
   *L. Goodman, Operations Engineer
   *S. Rafferty, Reactor Engineer
   M. Polsean, Shift Supervisor
   W. Nowicki, Supervisor, Instrument and Electric
   R. Wery, QA Supervisor
   *G. Joseph, Security Director
   *L. Kelley, Assistant Operations Supervisor
   *P. Shafer, Radiation Protection Engineer
   *B. Zibung, Health and Safety Supervisor
   *R. Brimer, Electrical Engineer
   D. Rybarik, Mechanical Engineer

   *Denotes those present at exit interview.

2. Operational Safety Verification

   The inspector observed control room operations, reviewed applicable logs and conducted discussions with control room operators during the period from December 22, 1981 through March 31, 1982. The inspector verified the operability of selected emergency systems, reviewed tagout records and verified proper return to service of affected components. The tours of the reactor buildings and turbine buildings 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 inspectors observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection controls. During the period from December 22, 1981 through March 31, 1982, the inspectors walked down the accessible portions of the Core Spray, Boron Injection, Shutdown Condenser and the No. 1A and 1B Emergency Diesel Generator Fuel Oil Supply 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.

   No items of noncompliance or deviations were identified.

3. Monthly Maintenance Observation

   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 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.

The following maintenance activities were observed/reviewed:

- Trouble shooting, cleaning and repairs to the 2400 volt switchgear (MR-00009)
- Repairs to the No. 1A Discharge Valve for the 1A Forced Recirculation Pump (MR-1088)
- Repairs to the No. 1B Emergency Core Spray Pump Breaker (MR-0895)

Following completion of maintenance on the 2400 volt offsite reserve breakers 252-RIA and 252-RIB and the repairs to the No. 1B Emergency Core Spray Pump Breaker, the inspectors verified that the systems had been properly returned to service.

No items of noncompliance or deviations were identified.

4. Monthly Surveillance Observation

The inspectors observed technical specifications required surveillance testing on the No. 1B Diesel Generator for the months of January and February 1982, 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 and 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 inspectors also witnessed or reviewed portions of the following test activities:

- Annual test of the Forced Circulation System Controls and automatically operated valves per Paragraph 3.7.1 of Chapter 3 of Volume 2 of the Operations Manual

- January monthly test of the No. 1A Diesel Generator
- Semiannual verification of CO_2 storage tank weight for the 1B Emergency Diesel Generator Room.

No items of noncompliance or deviations were identified.

5. IE Bulletin Followup

For the IE Bulletins listed below the inspector verified that the Bulletin was received by licensee management and reviewed for its applicability to the facility. If the Bulletin was applicable the inspector verified that the written response was within the time period stated in the Bulletin, that the written response included the information required to be reported, that the written response included adequate corrective action commitments based on information presented in the Bulletin and the licensee's response, that the licensee management forwarded copies of the written response to the appropriate onsite management representatives, that information discussed in the licensee's written response was accurate, and that corrective action taken by the licensee was as described in the written response.

(Open) IE Bulletins No. 79-26 and No. 79-26, Supplement 1 (Boron Loss from BWR Control Blades): A review of the licensee response letters (LAC-6726 dated January 9, 1980, and LAC-6919 dated May 15, 1980) indicated the licensee did not address their intentions to conduct the destructive test of a control blade as required by Section (4) of the Bulletin. The inspector discussed the omission with the licensee and determined that the licensee considered the Bulletin to not be applicable to LACBWR for the following reasons: (1) The Bulletin and its supplement discussed problems associated with General Electric (GE) control blades, LACBWR control blades were designed and manufactured by Allis-Chalmers; (2) The problems associated with the control blades discussed in the Bulletin and its supplement were caused by cracks in the stainless steel poison tubes, LACBWR control blades utilize inconel (Alloy 600) poison tubes. Although the licensee maintains that the Bulletin and its supplement are not applicable to LACBWR, the licensee has decided to conduct the destructive testing of a control blade as suggested in Section (4) of the Bulletin and to submit their findings to NRC Region III.

No items of noncompliance or deviations were identified.

6. 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.

- LER 81-09 (Nuclear Instrumentation Channel No. 7 High Scram Setpoints found to have drifted past Technical Specification limit)

- LER 81-10 (Mechanical Interlock on Personnel Airlock into Containment failed and allowed containment integrity to be breached while at power)

- LER 81-11 (No. 1B Diesel High Pressure Service Water Pump was made inoperable while radiator hose was replaced)

- LER 81-12 (All three Emergency Service Water Supply System Pumps were discovered to have less then the required fuel level as specified in Technical Specifications)

No items of noncompliance or deviations were identified.

7. <u>Plant Trips</u>

a. <u>Plant Scram on December 23, 1981</u>

Following the plant trip on December 23, 1981, the inspector ascertained the status of the reactor and safety systems by observation of control room indicators and discussions with licensee personnel concerning plant parameters, emergency system status and reactor coolant chemistry. The inspector verified the establishment of proper communications and reviewed the corrective actions taken by the licensee.

The following conditions developed which were not expected:

- There was a complete loss of offsite power due to the combined tripping of Oil Circuit Breaker 152 RI (Feed Breaker for the Reserve Transformer) and Circuit Breaker 252-RIB (1B Reserve Breaker). This loss of offsite power is discussed in greater detail in Licensee Reportable Occurrence No. 81-14.

- There was an unintentional cooldown of the No. 1B Forced Circulation (FC) Loop to below Technical Specification pressure/temperature limits. This cooldown was due to a combination of primary plant purification and excessive seal injection flow to the pump seals on the No. 1B FC Pump. This excessive cooldown of the No. 1B FC loop is discussed in greater detail in Licensee Reportable Occurrence No. 81-15.

USNRC Region III issued a Confirmatory Action Letter on December 28, 1981. This letter required the licensee to maintain the reactor shutdown until the cause had been determined for the above unexpected conditions and corrective actions had been taken to prevent recurrence.

The unintentional cooldown of the No. 1B FC loop was caused by a combination of equipment failure and personnel errors. The requirements listed below were violated during this event:

- Technical Specification 4.2.2.4.e states "The Forced Circulation loops shall not be pressurized unless their temperature is above 70°F and shall not be pressurized above 280 psig unless their temperature is at least 130°F."

- Technical Specification 6.8.1 states, in part, "Written procedures shall be established, implemented and maintained."

- Section 2.3.2 (Normal Plant Shutdown) Step 16 of Volume I of the Operations Manual states, in part, "Every 15 minutes during reactor cooldown record the "Reactor Vessel Wall Temperature Recorder" on the "Reactor Heatup and Cooldown Data Sheet." Do not exceed the limits of Section 2.1.3 Step 2." Section 2.1.3 Step 2 restricts cooldown rate at the forced circulation suction to 60°F/hr.

- Annunciator alarm response in Volume I of the Operations Manual, for Alarms E2-1 and E2-2 (FC Pump 1A and 1B Tripped) states, in part, under Immediate Action Step 3, "Ensure temperature in the shutdown loop remains greater than 200°F by reducing seal inject flow into the shutdown pump to minimum allowable D/P."

On December 24, 1981, while recovering from the reactor scram on December 23, 1981, plant personnel failed to take the required 15 minute reading for the No. 1A and 1B Forced Circulation (FC) loop suction temperature. This failure to take the required temperature readings contributed to personnel not noticing the excessive cooldown of the No. 1B FC loop that was caused by increasing rather than decreasing the seal injection flow rate to the No. 1B FC pump. These procedure violations resulted in the unintentional cooldown of the No. 1B FC loop to 86°F while loop pressure was greater than 280 psig. This is an item of noncompliance (409/81-23-01).

There was an attempt to return to power on January 21, 1982, but a reactor scram resulted when a wire to CRD No. 5 was broken while attempting to trouble-shoot the failure of CRD No. 5 to move. There was no record of a Red Phone notification to the NRC Operation Center of this event. It appears that the failure to report was an oversight and not intentional. This item was discussed with plant management and the licensee emphasized the necessity to report all automatic activitions of the reactor protective systems, regardless of how low the power level, to plant supervisors.

The plant was returned to operation on January 26, 1982.

b. Plant Scram on February 18, 1982

Following the plant scram on February 18, 1982, the inspector ascertained the status of the reactor and safety systems by observation of control room indicators and discussions with licensee personnel concerning plant parameters, emergency system status and reactor coolant chemistry. The inspector verified the establishment of proper communications and reviewed the corrective actions taken by the licensee.

All systems responded as expected and the plant was returned to operation on February 18, 1982.

No items of noncompliance or deviations were identified.

c. Plant Scram on March 30, 1982

Following the plant scram on March 30, 1982, the inspector ascertained the status of the reactor and safety systems by observation of control room indicators and discussions with licensee personnel concerning plant parameters, emergency system status and reactor coolant chemistry. The inspector verified the establishment of proper communications and reviewed the corrective actions taken by the licensee.

All systems responded as expected and the plant was returned to operation on March 31, 1982.

No items of noncompliance or deviations were identified.

8. Followup on Open Inspection Items

a. (Closed) Unresolved Item (409/81-07-02): Loss of feedwater heating transient of March 16, 1981.

While operating at 85% power on March 16, 1981, a problem was experienced with sticking of the turbine generator governor valve linkage (secondary relay valve). The sticking resulted in fluctuations of reactor coolant system pressure which in turn caused the main steam bypass valve to open to 34% to maintain reactor pressure. The licensee attempted several times to free the linkage by using the instructions contained in Operating Memorandum DPC-86, Revision 0, to move the load limiter up and down and thereby exercise the governor valves and their linkage. When this failed, the turbine generator was taken off the line (governor and stop valves closed) and the 85% steam flow was automatically diverted to the main condenser via the main steam bypass valve. The plant remained in that condition for approximately 12 minutes until the linkage problem was corrected and the turbine generator governor and stop valves reopened.

The inspectors reviewed this event utilizing the additional information outlined in Inspection Report No. 50-409/81-21 and new information provided by the licensee, including all docketed safety analyses that address the subject. The four significant areas of concern outlined in Inspection Report No. 50-409/81-21 are readdressed below to reflect all new information obtained.

(1) Safety Analyses

The act of taking the turbine off the line and bypassing the 85% steam flow to the main condenser effectively resulted in loss of all three feedwater heaters. The heaters use turbine extraction steam as the principal source of heat for the feedwater and with the turbine governor and stop valves closed there is no extraction steam. The result is a decrease in the temperature of the feedwater being pumped into the reactor vessel which constitutes a positive reactivity insertion.

The inspector's reviewed the below listed docketed analyses to determine if the analyses bound the conditions that occurred during the March 16, 1981 event:

- The docketed analysis for Turbine Trip With No Scram in itself did not bound the event because of the short time duration of the analysis (i.e., 40 seconds for the entire transient) and because the affect of feedwater temperature change was not addressed.

- The analysis for the Loss of a Single Feedwater Heater indicated that the loss of a single feedwater heater at 100% power would result in a 70°F drop in feedwater temperature and a power increase to 116% (assuming no scram) but there would be no core damage. The operation on March 16, 1981, at 85% power resulted in a 150°F decrease in feedwater temperature and a power increase to about 102% with no apparent core damage. Since the initial reactor power level was less than 100%, the decrease in feedwater temperature by 150°F resulted in actual consequences which were less severe than the transient analyzed.

- The analysis of Main Steam Bypass Valve Opening at Power did not bound the March 16, 1981, event because the analysis indicated that reactor pressure would drop causing a decrease in reactor power whereas on March 16 the reactor pressure remained approximately constant and core power increased.

- The analysis for Load Loss from 55% Power did not bound the March 16, 1981, event because a load loss at power would not result in the total closure of the turbine stop valve as occurred on March 16 and because the analysis only monitored the event for the first 70 seconds and did not consider the power increase caused by the drop in feedwater temperature.

The inspectors concluded that the event of March 16, 1981, was bounded by a combination of the Turbine Trip With No Scram and Loss of a Single Feedwater Heater analyses. The core power increase experienced on March 16, 1981, was nearly the same as the core power increase predicted in the safety analysis (17% vs. 16%).

It is the NRC position that the licensee must submit a request to the NRC for review of this mode of operation prior to any future planned operations involving the use of the main steam bypass valve while at power. The basis for this determination is that (1) the operation on the main steam bypass valve with two feed pumps in operation may an unreviewed safety question, and (2) the nuclear instrumentation is incapable of accurately detecting actual core power as required by the Technical Specifications during this mode of operation.

(2) Instrumentation

During the bypass operation, the actual reactor power increased from 85% to approximately 102% as subsequently determined from heat balance data. However, during the event the power range nuclear instrument indication only increased to 85% power. The automatic high power scram setpoints during this event were nonconservative because the instruments were not indicating true power. The intrumerts were indicating approximately 15% below actual core power. The inability of the instruments to accurately reflect true power while operating at power on the main steam bypass valve results in the inability during such operation to satisfy Technical Specification 2.10.3.3 requirements to detect and indicate power levels and initiate scram actions at special power levels. This is considered to be part of an item of noncompliance (409/81-23-02A). The licensee has prohibited operation in the steam bypass mode during power operation.

(3) Operating Memoranda

On March 17, 1981, the licensee issued Revision 1 of Operations Memorandum DPC-86 to allow plant operation on the main steam bypass valve as necessary to correct governor valve linkage problems. Following discussions with the inspectors, the licensee recognized that the information contained in Revision 1 should have been issued in the form of a procedure. Therefore, Revision 2 was issued on April 22, 1981, to rescind the provisions of Revision 1. The rescinded information was then incorporated into a change to annunciator response Procedure B14-3, "Main Steam Bypass Valve Not Closed," contained in Section 3.2 of Volume 1 of the Operations Manual.

The inspectors reviewed the licensee's Operating Memoranda system and noted that there were about 15 effective memoranda. The inspectors determined that some of the memoranda address matters which would normally be addressed in procedures. In addition to DPC-86, Revision 1, other examples of memoranda which appear to require procedures are DPC-78, 80, 81, 83, 84, and 91. The inspectors' concern is that the memoranda approval chain is not as stringent as the approval chain for procedures in that approval by the Operations Review Committee (ORC) is not required. Thus, for those cases where Operations Memoranda are issued in lieu of procedures, the Technical Specification 6.8.2 requirement to have procedures reviewed by the ORC is circumvented.

The failure to perform a safety evaluation or have the ORC review Operations Memorandum DPC-86 prior to implementation was in violation (409/81-23-03) of the requirements listed below:

- 10 CFR 50.59 states, in part, "(a)(1) The holder of a license authorizing operation of a production or utilization facility may...(ii) make changes in the procedures as described in the safety analysis report... without prior Commission approval, unless the proposed change...involves a change in the Technical Specifications incorporated in the license or an unreviewed safety question...(b) The licensee shall maintain records of... changes in procedures made pursuant to this section... These records shall include a written safety evaluation which provides the bases for the determination that the change...does not involve an unreviewed safety question..." Sections 8.3 and 13.4 of the safety analysis report specify the operational uses of the main steam bypass valve and those uses do not include use of the valve to bypass the total steam flow directly to the condenser during high power operation.

- Procedures and changes thereto that are required for the operation of the main steam system are among those procedures which are required by Technical Specification 6.8.2 to be reviewed by the Operation Review Committee prior to implementation.

The inspector reviewed the status of the commitment made by the licensee to conduct a review of all Operations Memoranda and plant policy covering the issuance of Operations Memoranda. The inspectors' review indicated that an effort to review and cancel outdated and unneeded Operations Memoranda was in progress by the licensee. All existing Operations Memoranda will be cancelled prior to the end of the 1982 refueling outage. There has not been a written policy established to regulate the issuance of Operations Memoranda. This item will be reviewed during a subsequent inspection.

(4) Preplanning and Operator Awareness

The apparent reason for the March 16, 1981, operation was to avoid a possible scram due to reactor pressure fluctuations and to avoid a shutdown to correct the linkage problem. Bot the scram and orderly shutdown would have entailed a loss of electrical generation during plant recovery and the performance of radioactivity analysis required by the Technical Specifications. Therefore, for operational convenience during a plant transient the temporary total bypass steam flow operation at 85% power was conducted.

This operation was not in accordance with existing plant procedures. Technical Specification 6.8.1 requires, in part, that written procedures be implemented covering activities referenced in Appendix A of Regulatory Guide 1.33, Revision 2. Appendix A of Regulatory Guide 1.33, Revision 2, references in Paragraphs 4.1 and 4.m operation of the main steam system. Plant procedures contained in Sections 2.1.3, 2.3.2, 3.3.1.3 of Volume I of the Operations Manual and response procedures for Alarms A1-3, B14-3 and D2-3 provide for the use of the main steam bypass valve during normal plant startups and shutdowns and during times when the reactor coolant system pressure is increasing above an established value but indicate that the valve is normally closed during power operation. (These plant procedures are consistent with the procedures and system descriptions contained in the safety analysis report.) Neither the plant procedures nor the safety analysis report indicate that the main steam bypass valve will be used to bypass all of the generated steam directly to the condenser during power operation. This failure to follow existing procedures is part of an item of noncompliance (406/81-23-02B).

Information reviewed by the inspector indicated a lack of preplanning for this significant plant evolution as evidenced by the failure of supervisory operating personnel to recognize the need for and to obtain a properly approved procedure. In the absence of such a procedure, the proper course of action would have been to conduct an orderly plant shutdown.

The licensee committed to reinstruct operating personnel and to ensure the operator training and retraining program is augmented as necessary to ensure a higher degree of operator awareness of plant transients. This commitment was verified by the inspector to be completed. This should ensure a higher degree of operator awareness of plant transients.

b. (Closed) Noncompliance (409/81-14-01): Failure of plant staff to follow Special Work Permit (SWP) procedure Section 6.5.3 of Volume I of Operations Manual.

In their response letter (LAC-7857 dated October 9, 1981), the licensee indicated that the contributing factors to this incident were the random frequencies of involvement of management personnel in hands-on maintenance activities of this nature, the relative obscurity of the posting of the job site copy of the SWP, and the attentiveness of the technician to other job-site tasks.

The licensee determined that procedural changes could be made to ensure uniformity in individual dose logging requirements. Chief among these changes was the implementation of a revised procedure which adopted a 4-color coded SWP form and the establishment of a prominent, convenient, central location for posting the central access point copy of the SWP and the dosimeter log form. The procedure revision was accomplished on June 24, 1981. The central access posting facility was established a few weeks later in the Change Room which is the common entrance and exit for plant radiation control areas.

The inspector has observed implementation of this new method of (SWP) conttrol and the licensee's corrective action appears to have corrected the cause of the procedure violation.

c. (Open) Unresolved item (409/81-06-4): Additional Accident Monitoring for Containment Pressure and Level, TMI Items II.F.1.4 and II.F.1.5.

The licensee has submitted additional information to NRR (LAC 7990 dated December 23, 1981). This item will remain open until a response from NRR is received.

d. (Closed) Unresolved Item (409/81-02-03): Guidance on Procedures for Verifying Correct Performance of Operating Activities, TMI Item I.C.6.

The licensee did not submit a request to NRR for exception to the requirements as stated in NUREG-0660 and later clarified by NUREG-0737. The licensee's implementation procedures (ACP 15.2, "Equipment Lock and Tag Control," and ACP 17.3, "Maintenance Requests") for the requirements of NUREG-0737 applied the requirement for Verification of Correct Performance of Operating Activities only to maintenance and did not address the requirements of this Verification for Surveillance Testing. The licensee has developed a list of those systems considered important to safety utilizing the introduction paragraph to Appendix A of 10 CFR 50 as a definition of "important to safety." Subsequently, the licensee has modified the ACPs to include Surveillance Testing in addition to maintenance within the scope of their program for Verification of Operating Activity.

Based on the above considerations, the licensee has met the requirements as specified in Item I.C.6 of NUREG-0737. Verification of the adequacy of the licensee's procedure changes has been completed.

e. (Closed) Unresolved Item (409/81-06-05): Modification of RCIC System, TMI Item II.K.3(22).

This item was declared to not be applicable by LACBWR (LAC 7112 dated September 3, 1980) due to the fact that LACBWR does not have a RCIC system. The inspector and NRR Project Manager agree that the item is not applicable to LACBWR.

9. <u>TMI Task Action Plan</u>

a. The inspector verified that the licensee took appropriate action to meet written commitments to NRR for the following TMI Task items.

(1) (Closed) (I.A.1.3.1) Shift Manning, Limit of Overtime.

The licensee issued new Administrative Control Procedure ACP-2-9, "Overtime Policy," to specifically set forth a policy concerning overtime requirements as delineated in NUREG-0737 and NRR letter of July 31, 1980, exception was taken for requirements specified for the STA. The ACP only limits overtime for SROs, ROs, and supervisory personnel holding SRO licenses when performing licensed duties.

The STAs spend 24 continuous hours at the facility approximately every 8-10 days depending upon the number of qualified STAs available. Sleeping quarters are provided for STAs. The ACP states that the hours the STA spends onsite outside of normal working hours are considered on-call hours. The STA is required to be onsite and able to respond to the Control Room within ten minutes, but is not required to routinely work overtime. During fuel handling, the STA may perform the fuel accountability functions. The hours the STA is required to routinely work fall within the limitations of NUREG-0737.

The licensee meets the requirements for administrative procedures to include overtime restrictions as delineated in the NRR letter of July 31, 1980.

b. (Open) (II.F.2.3.B) Instrumentation for Detection of Inadequate Core Cooling.

The licensee indicated in LAC 6769 dated January 31, 1980, that present equipment was adequate. In LAC 6853 dated April 9, 1980, the licensee answered the NRR question about an ongoing program to stay abreast of any new level equipment. NRR's April 25, 1982, letter left the item open until NRR completes its review. This item is unresolved (409/81-23-04) and will be reviewed during a subsequent inspection.

c. (Closed) (II.K.3.14) Isolation Condenser Isolation Modification.

The licensee indicated in LAC 7112 dated September 3, 1980, that modification was not required. NRR accepted this position in a December 12, 1981, letter.

d. (Closed) (II.K.3.24) Space Cooling for HPCI/RCIC modification.

This item was declared to not be applicable by the licensee in LAC 7112 dated September 3, 1980, due to the fact that LACBWR does not have a RCIC or HPCI system.

e. (Open) (II.K 3.25.B) Effects of Loss of Alternating Current Power on Pump Seals.

The licensee indicated in LAC 7684 dated July 22, 1981, that no modifications are necessary. This item is still under review by NRR. This matter is unresolved (409/81-23-05) and will be reviewed during a subsequent inspection.

f. (Open) (II.K.3.28) Qualification of ADS Accumulators.

The licensee submitted their evaluation to NRR for review as required by NUREG-0737. The inspector verified that the licensee accomplished this task by the January 1, 1982, due date and that appropriate information was contained in the licensee's letter LAC 7990 dated December 23, 1981. This item is unresolved (409/81-23-06) until NRR completes its review and will be reviewed during a subsequent inspection.

10. Training and Requalification Training

The inspector reviewed the licensee training and requalification training program as described in Administrative Control Procedures (ACP) 21.1, 21.2, 21.3 and 23.1 to verify that the programs conform to Technical Specifications and TMI requirements. Individual training records of a variety of employees were examined to determine that the required training was being conducted and that required records were being maintained. The inspector interviewed several employees to verify that the training was actually being conducted and the records of attendance accurately reflected the actual attendance.

The following inspector findings were identified to the licensee for resolution:

a. Administrative Control Procedure ACP 23.1 "LACBWR Crane Operator Qualification and Certification" was changed and the added information was not provided to the qualified crane operators on a timely basis.

b. Operator requalification tests did not include the date that the tests were taken.

c. Several of the lectures attended by persons in the operator requalification program were not promptly recorded in their personnel records.

d. The personnel records of persons in the operator requalification training program do not include documentation that those simulated control manipulations (required by ACP 21.2 Step 6.1) are accomplished. Also, the licensee requalification program does not specify that those simulated control manipulations should be accomplished in the physical area where the operator would normally expect to encounter the conditions simulated. Past practice has been to talk-through rather than walk-through the simulated manipulations.

e. After discussion with several operators enrolled in the requalification program, the inspector ascertained that the quizses given after a specific lecture are administrated on an informal basis with the lecturer giving the quizzes to the attendees to complete and turn in whenever they can. This item was discussed with Mr. Joseph McMillen (Chief Operator Licensing Branch, Region III). Mr. McMillen indicated that this was not a good practice and should not continue.

The above items are unresolved items (409/81-23-07) and will be reviewed during a subsequent inspection.

11. Type B and C Containment Leak Rate Testing

The inspector was notified on March 14, 1982, by the Plant Superintendent that the Type B test of the personnel airlock, accomplished March 13, 1982, was determined to be unsatisfactory based on a review of the leak rate data. The Plant Superintendent indicated that it was suspected that the leak rate was not indicative of the actual leak rate due to the fact that the operator who conducted the test believed he had read a gage improperly. The inspector, after consulting Region III management, suggested that the plant conduct an orderly shutdown and in parallel with the shutdown, the licensee should reperform the test.

The inspector witnessed portions of the repeated test and independently verified the licensee's results. This retest confirmed the Plant Superintendent's suspicion that the actual leak rate was acceptable and below that value specified in Technical Specification 5.2.1.2(b)(1). The plant shutdown was terminated and power escalation was commenced. The inspector suggested that the licensee review the procedure specified in Section 3.5.1 of Volume XI of the Operations Manual and determine if additional instructions are needed.

No items of noncompliance or deviations were identified.

12. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. Unresolved items disclosed during inspection are discussed in Paragraphs 9 and 10.

13. Exit Interview

The inspector met with licensee representatives (denoted in Paragraph 1) throughout the month and at the conclusion of the inspection and summarized the scope and findings of the inspection activities.