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

REGION III

Report of Operations Inspection

IE Inspection Report No. 050-155/76-13

Licensee: Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Big Rock Point Nuclear Plant
Charlevoix, Michigan

License No. DPR-6
Category: C

Type of Licensee: BWR GE 240 MWt

Type of Inspection: Routine, Announced

Dates of Inspection: June 18-19 and 21-24, 1976

Principal Inspector: for E L Jordan D. E. Hunter 7/8/76 (Date)

Accompanying Inspector: None

Other Accompanying Personnel: None

Reviewed By: E L Jordan E. L. Jordan, Chief
Reactor Projects Section 2
7/8/76 (Date)

8101210321

SUMMARY OF FINDINGS

Inspection Summary

Inspection on June 18-19 and 21-24, 1976, (76-13): Review of preoperational testing, startup testing, operations, reportable occurrence, headquarters requested item and selected outstanding items. One item of noncompliance was identified concerning failure to perform temporary procedure changes in accordance with the Technical Specifications.

Enforcement Items

The following item was identified during the inspection:

A. Infraction

Contrary to Technical Specification 6.8.3, temporary procedure changes regarding containment entry during the Hot Valve Operability Test were made on June 17 and 18, 1976 without the required approvals. (Paragraph 3, Report Details)

Licensee Action on Previously Identified Enforcement Items

A review of plant modification controls indicates that the licensee's corrective actions are not complete. (Paragraph 10, Report Details)

Other Significant Findings

A. Systems and Components

1. The startup testing of the reactor depressurization system has been delayed after problems were encountered with operation of the depressurization valves. (Paragraph 3, Report Details)

2. Unresolved Item: The lack of documentation at the site of a design review concerning the modification to the emergency diesel generator control circuit is considered an unresolved item pending further review. (Paragraph 8.g., Report Details)

B. Facility Items (Plans and Procedures)

The low power physics testing has been completed following the initial criticality on June 16, 1976.

C. Managerial Items

None.

D. Noncompliance Identified and Corrected by Licensee

None.

E. Deviations

None.

F. Status of Previously Reported Unresolved Items

1. The design review concerning the electrical power supplies for the reactor depressurization system uninterruptable power supplies has been completed. This item is considered resolved. (Paragraph 5.a., Report Details)

2. The authorized inspector sign off of the welds in the ring core spray line has been completed. This item is considered resolved. (Paragraph 5.b., Report Details)

3. The authorized inspector sign off of the 1974 inservice inspection has been completed. This item is considered resolved. (Paragraph 5.c., Report Details)

Management Interview

The management interview was conducted on June 24, 1976, by Mr. Hunter with the following persons present:

R. B. DeWitt, Manager, Production
C. J. Hartman, Plant Superintendent
D. E. DeMoor, Technical Engineer
C. R. Abel, Operations Superintendent
T. W. Elward, Technical Superintendent
J. P. Flynn, Maintenance Superintendent
G. B. Szczotka, Quality Assurance Superintendent
A. C. Sevener, Operations Supervisor
R. W. Voll, Reactor Engineer
V. A. Avery, Shift Supervisor

A. The inspector stated that a review of operations revealed only one questionable item concerning the procedural controls for rod withdrawal and insertion in single notch step sequences. The operating procedure and the Technical Data Book did not appear to provide clear instructions for the operators.

The licensee stated that the area would be reviewed and the appropriate corrective actions taken. (Paragraph 2.e., Report Details)

B. The inspector stated that containment entries made during the performance of STP-10 on June 17 and 18, 1976, which were contrary to the hot valve operability procedure represented noncompliance with Technical Specification 6.8.3. which requires approvals of temporary procedure changes prior to implementation. The licensee stated that temporary procedure changes had been subsequently issued and a deviation report had been issued and reviewed concerning the failure to follow procedures. The licensee stated that the failure to make the procedure change was an oversight.

The inspector asked the licensee to insure that all plant departments were made aware of this problem. The licensee stated that the appropriate plant staff members would be informed. No further response is required concerning this item of noncompliance. (Paragraph 3, Report Details)

C. The inspector stated that a review of procedure D2.25, Emergency Shutdown, revealed no subsequent operator actions to monitor plant conditions.

The licensee acknowledged the statement and noted that considering all equipment operating with no assumed failures, the reactor system would reach a cooled condition via the normal operation of the emergency condenser system. The inspector stated that this matter would be reviewed further during a subsequent inspection. (Paragraph 4, Report Details)

D. The inspector stated that a review of the open item concerning the uninterruptable power supply battery specific gravities and cell replacement was verified to be completed. The inspector stated that he had noted that the "C" and "D" batteries continued to exhibit specific gravity readings at the lower end of the acceptance criteria. The licensee acknowledged the statement by the inspector. (Paragraph 6, Report Details)

E. The inspector stated that a review of the facility change package concerning the reactor depressurization system electrical power supplies revealed no discrepancies. This item is considered closed. The licensee acknowledged the statement by the inspector. (Paragraph 5.a., Report Details)

F. The inspector stated that a review of the facility change concerning the modification of the emergency diesel generator control circuit revealed that no detailed design review was available at the site at the time of the inspection.

The licensee stated that the design review was being located and will be provided. The inspector stated that this type of inadequacy must be prevented in the future with particular attention being given to facility changes until the design control procedures are fully implemented.

During a subsequent telecon with the licensee on June 26, 1976, the inspector stated that this item will be carried as unresolved pending location of the design review onsite to support the documented safety evaluation. (Paragraph 8.g., Report Details)

REPORT DETAILS

1. Persons Contacted

C. J. Hartman, Plant Superintendent
D. E. DeMoor, Technical Engineer
C. R. Abel, Operations Superintendent
T. W. Elward, Technical Superintendent
J. P. Flynn, Maintenance Superintendent
G. B. Szczotka, Quality Assurance Superintendent
R. W. Voll, Reactor Engineer
A. C. Sevener, Operations Supervisor
R. W. Doan, Shift Supervisor, Training Coordinator
E. F. Peltier, Shift Supervisor
S. A. Carlisle, Shift Supervisor
R. A. Curtis, Control Room Operator
H. E. Downing, Control Room Operator
D. J. Horstman, Control Room Operator
W. J. Woods, Control Room Operator
J. L. Kuemin, Plant Engineer
D. D. Herboldsheimer, Maintenance Scheduler
H. M. Phelps, Assistant I&C Supervisor
G. H. Petitjean, Plant Engineer
S. E. Martin, Project Engineer
W. Clark, Projects Construction Superintendent
K. F. Krueger, Consumers Startup Engineer
A. J. DeGrasse, Catalytic Startup Coordinator

2. Review of Operations

The inspector reviewed the following selected records of routine plant operations and conducted plant tours to verify activities to be in accordance with Technical Specifications and Administrative Procedures:

a. Shift Supervisor Log, May 31, 1976, through June 18, 1976.
b. Control Room Log, May 8, 1976, through June 18, 1976.
c. Reactor Operations Log, April 27, 1976, through June 20, 1976.
d. Control Room Data Sheets, May 5, 1976, through June 23, 1976.
e. Operating Memos, May 26, 1976, through June 17, 1976.
f. Control Room Status Board.
g. Outstanding Tagging and Tagging Orders.
h. Fuel Status Boards.
i. Equipment Rotation.
j. Plant Annunciators.

k. The position of selected valve positions on the fire protection system including the supply valves to the core spray system, emergency makeup to the main condenser and the hose connection for the core spray recirculation heat exchanger alternate cooling water supply were examined during a plant tour.

l. Technical Data Book

The review of the Technical Data Book, Technical Specification 5.2.6, and Operating Procedures, Sections B1.1.6 and B1 3.3.2, concerning control rod withdrawal and insertion sequencir quirements indicated a procedural weakness. The insp rified through observation of rod movement by oper rs and discussions with operators that rods are moved in single steps within the groups following a mirror image pattern with no rod more than one step apart. The procedural steps do not appear to be definitive enough to prevent an operator deviation from the rod sequence within a specific group of rods.

m. Operational Surveillance Tests

The inspector reviewed selected surveillance tests performed during the outage. No discrepancies were noted.

TR-06 - Liquid Poison System Check Valve Test, performed on February 11, 1976.
TR-08 (365-05) - Core Spray System Check Valve Test.
TR-09 - Core Spray Heat Exhanger Shell Side Flow, performed on June 1, 1976.

This test was performed to verify adequate flow through the temporary hose connection between the fire protection system test header and the CS heat exhanger. The minimum required flow was 131,425 lbs/hr and the conservative calculated flow was 162,000 lbs/hr at a fire pump discharge pressure of 140 psig.

TR-16 - Emergency Diesel Generator Auto Start, performed on April 8, 1976.
TV-10, Rev. 2 - Hydrostatic Test of NSSS, performed on June 14, 1976.

n. Startup Checklists

The inspector reviewed selected startup checklists and valve checkoff lists, including the Master Checklist, incore instruments, neutron monitoring systems, condensate system, control rod drive system, post-incident system, emergency condenser system, fire protection system and plant locked valves.

No discrepancies were noted.

3. Reactor Depressurization Startup Testing of Program

The inspector reviewed the procedure for startup testing of the reactor depressurization system (RDS) and made direct observations of the plant conditions during portions of the testing.

The O-RDS-1 (STP-10), Rev. 2, Hot Valve Operability Test, was performed partially on June 16-18, 1976, at 100 psig and 700 psig. The licensee found that the RDS valves leaking through slightly and the isolation valve open limit switches were damaged due to apparent increased travel at higher system pressure. The test was terminated on June 18, 1976, and the plant was placed in the cold shutdown condition. The procedure was evaluated and revised to allow testing at 1350 psig to provide adequate pressure to lift the RDS valves via the bypass line.

The inspector noted on June 18, 1976, that maintenance and inspection was being performed inside he containment vessel with pressure in the reactor vessel. These activities were not in accordance with the procedure, step 2.2, which stated that "No entries will be made while there is pressure in the reactor vessel, except to adjust CA-135. Entry under this condition will be with the power to the valves tagged off at their respective breakers." Record review revealed that an entry had also been made on June 17, 1976, for maintenance purposes. No temporary procedure changes were issued to provide immediate evaluation of the activities.

Failure to provide the required temporary procedure changes is considered an item of noncompliance pursuant to Technical Specification 6.8.3.

Following identification by the inspector, the licensee issued temporary procedure changes concerning the deviation from the procedure and issued a deviation form (QA-16) concerning the failure to provide the temporary procedure changes as required by the Administrative Procedures.

The licensee recommenced the startup testing on June 20, 1976, and heated the plant to approximately 1200 psig in accordance with procedure O-RDS-1 (STP-10), Rev. 4. During the test at approximately 500 psig and 1200 psig, problems were encountered with the Target Rock Valves. The "A" and "C" valve opened and closed normally upon demand. The "B" and "D" valves malfunctioned, failing to open upon demand and failing to close upon demand, respectively. The plant was placed in the cold shutdown condition to await an evaluation and resolution of the valve malfunctions.

4. Emergency Shutdown Procedure D2.25

The inspector reviewed the procedure to provide instructions for shutdown of the plant from outside the control room. The procedure required the operator to proceed to the No. 1 and No. 2 reactor protection buses and open the protection breakers (CB 40A1 and CB 40A2). The procedure does not indicate any subsequent operator actions to monitor the plant conditions following the remote trip, but indicates that the emergency condenser will go into service.

5. Previously Unresolved Items

The inspector reviewed previously unresolved items to verify corrective action completion by the licensee.

a. The design review of facility change FC-351 concerning the electrical power load addition to the 1A and 2A electrical buses for the reactor depressurization system was reviewed.[1/]

The design review was performed by the licensee and the package included the bus load requirements and breaker tripping requirements.

No further questions are required at this time and this item is considered resolved.

b. Core Spray We[illegible]

The revie[illegible] the core spray weld (North, 1A and 14A) packages[2/][3/] indicated that the authorized inspector reviewed and signed off on May 6, 1976.

This item is considered resolved.

c. Inservice Inspection

The review of the 1974 inservice inspection[4/][5/] indicated that the authorized inspector signed off on June 14, 1976.

This item is considered resolved.

1/ IE Inspection Rpt No. 050-155/76-12.
2/ IE Inspection Rpt No. 050-155/76-09.
3/ Ltr, CP to IE:III, dtd 5/26/76.
4/ IE Inspection Rpt No. 050-155/75-13.
5/ IE Inspection Rpt No. 050-155/76-10.

6. Outstanding Items

a. The inspector's review of the startup test procedure package, STP-011,[6] UPS Functional Test, revealed that the specific gravities for all of the battery cells had been taken and recorded, the UPS battery cells D-3 and D-20 had been replaced and all of the cell specific gravity readings indicated 1.200 or greater. The inspector noted that the "C" and "D" batteries continued to exhibit specific gravity readings at the lower end of the acceptance criteria.

   No further questions remain at this time and this item is considered closed.

b. The inspector's review of the emergency procedure, training and walk throughs,[7] indicated that the required training was completed and the shift supervisors and operators who had missed the training were being updated prior to assuming shift responsibilities.

   No further questions are required at this time and this item is considered closed.

c. The inspector's review of the operations training subject material covered during the outage to update the operators concerning the RDS modification, ECCS modifications and other essential facility and procedure changes appeared adequate. The shift supervisors and operators who missed the training were being updated prior to assuming shift responsibilities.

d. The inspector's review of the use of the Functionally Equivalent Substitution (FES) memos[8] revealed that the Plant Review Committee had reviewed the memos. The Plant Superintendent has assigned the task of writing an Administrative Procedure to control the use of the FES memos.

   This item will remain open pending completion and review of the completed Administrative Procedure.

6/ IE Inspection Rpt No. 050-155/76-12.
7/ IE Inspection Rpt No. 050-155/76-07.
8/ IE Inspection Rpt No. 050-155/76-10.

e. The inspector's review of the control of the isolation valve[9/][10/] for the differential pressure switches (dps/9051 and dps/9052) and the vacuum indicator (PT-173) indicated that the valve will be normally closed during normal plant conditions and will be unisolated following a LOCA as the pressure decreases below 5 psig and reisolated any time the containment pressure exceeds 8 psig. The instructions for operation of the root valve located at penetration H-96 are included in step D3.3.2.4 and Appendix B to D3.3 (Loss of Reactor Coolant).

No further questions are required at this time and this item is considered closed.

f. The inspector's review of the surveillance procedure, for the control and indication slow blow fuses,[11/][12/] revealed that the surveillance test (T30-23, Rev. 0, 6/7/76) is scheduled to be performed monthly.

The inspector verified that the fuse blocks were labeled and that the operator on duty was familiar with the fuses and the surveillance test. This item is considered closed.

7. Startup Testing After Refueling

The inspector reviewed selected startup procedures to insure selected tests were performed in accordance with the Technical Specifications.

a. TR-21 - Control Rod Drive Friction Testing Procedure, performed on February 6, 1976.

b. TR-46 - Core Load Procedure, performed on May 1, 1976.

c. TR-43 - Shutdown Margin Check, performed on May 3, 1976.

The inspector verified that the test would be terminated if the next predicted rod movement would take the reactor critical.[13/]

For Cycle 14, the single rod stuck shutdown margin test was performed satisfactorily.

d. TR-44 - Moderator Temperature Coefficient, performed on June 17, 1976.

No discrepancies were noted.

9/ RO 050-155/01-76.
10/ IE Inspection Rpt No. 050-155/76-04.
11/ IE Inspection Rpt No. 050-155/76-12.
12/ Ltr, CP to NRR, dtd 5/20/76.
13/ IE Inspection Rpt No. 050-155/75-11.

8. Facility Changes

The inspector reviewed selected facility changes to insure they were performed in accordance with the Administrative Procedures, Technical Specifications and 10 CFR 50.59.

a. CIS-76-FC-349 - Addition of the resin sluice line manual isolation valve and upgrading the resin sluice line valves.14/15/

No discrepancies were noted.

b. CIS-76-FC-328 - Addition of a tell-tale drain on the resin sluice line.16/

No discrepancies were noted.

c. SPS-76-FC-359 - Addition of RDS control panel power supply to panel 1Y.

No discrepancies were noted.

d. SPS-76-FC-338 - Bus 2B extension.

No discrepancies were noted.

e. PIS-76-C-376 - Addition of core spray flow recorder on the control panel.

No discrepancies were noted.

f. PSI-76-C-377 - Providing electrical switching for core spray flow to the recorder on the control panel.

No discrepancies were noted.

g. SPS-76-C-358 - Providing 125V DC for the emergency diesel generator control circuit from the RDS uninterruptable power supply "A".

The review of the facility change revealed that the offsite design review was not contained within the package. A memorandum from offsite to the project engineer indicated that the design review had been performed.

The lack of a documented design review at the site will be carried as unresolved pending arrival and review of the design review documentation onsite to support the documented safety evaluation.

14/ AO 050-155/19-75.
15/ IE Inspection Rpt No. 050-155/75-15.
16/ AO 050-155/21-75

9. Reportable Occurrence

The inspector reviewed the following reportable occurrence to assure adequate review, evaluation and reporting.

LER RO-9-76 - Failure of the Emergancy Diesel Generator Breaker to Close Upon Loss of Power

The licensee reported[17/] that the emergency diesel generator breaker failed to close upon loss of power to the 2B bus. The inspector reviewed the event with the licensee's representative and verified that subsequent inspection and testing of the emergency diesel generator breaker revealed no discrepancies. The breaker and circuit has performed satisfactorily subsequently.

No discrepancies were noted.

17/ Ltr, CP to IE:III, dtd 6/9/76.