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

REGION III

Report No. 50-155/77-17

Docket No. 50-155 License No. DPR-6

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

Facility Name: Big Rock Point Nuclear Plant

Inspection At: Big Rock Point Site, Charlevoix, MI

Inspection Conducted: November 15-18, 28, 29, and December 1 and 2, 1977

D. R. Hunter

Inspector: D. R. Hunter 12/29/77

RFWarnick

Approved By: R. F. Warnick, Chief 12/29/77
Reactor Projects Section 2

Inspection Summary

Inspection on November 15-18, 28, 29, and December 1 and 2, 1977 (Report No. 50-155/77-17)
Areas Inspected: Routine, unannounced inspection of plant operations; refueling operations; startup operations; reportable occurrences; IE Bulletins and Circulars; items of noncompliance; quality assurance items; records; safety limits, limiting safety system setpoints, and limiting conditions for operation; and outstanding inspection items. The inspection involved 59 inspector-hours onsite by one NRC inspector.
Results: Of the ten areas inspected, no apparent items of noncompliance were identified in eight areas; three apparent items of noncompliance were identified in one area (deficiency - failure to follow procedures - Paragraph 3.f; deficiency - failure to maintain records - Paragraph 3.f; deficiency - unauthorized procedure change - Paragraph 3.k.)

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DETAILS

1. Persons Contacted

*C. J. Hartman, Plant Superintendent
*J. P. Flynn, Operations Superintendent
*J. A. Rang, Maintenance Superintendent
*D. E. DeMoor, Technical Engineer
*R. E. Schrader, Technical Superintendent
A. C. Sevener, Operations Supervisor
*C. R. Abel, Senior Engineer
*R. H. Brzezinski, I&C Engineer
*D. P. Blanchard, Reactor Engineer
F. Valade, Shift Supervisor
S. Carlisle, Shift Supervisor
T. Pence, Shift Supervisor
E. Peltier, Shift Supervisor
*G. Gilbody, QA Engineer
J. A. Johnson, I&C Supervisor
J. J. Popa, Maintenance Engineer
*K. A. Brun, PRC Secretary

The inspector contacted several other licensee employees, including members of the technical, administrative, and operations groups.

*denotes those attending the management exit on November 18 and December 2, 1977.

2. Licensee Action on Previous Inspection Findings

(Open) Noncompliance (50-155/77-08): The procedure for calibration of the incore detectors has not been completed.

(Closed) Unresolved Item (50-155/77-08): The completion of the instrumentation operability checklists (C-2 and C-3) verifying channel operability during the shutdown margin testing (BRP-RE-08) on May 3, 1976, was not apparent and is an item of noncompliance pursuant to Technical Specification 6.10.1 and is a deficiency. (Paragraph 3.f)

(Closed) Open Item (50-155/77-08): The licensee provided an error analysis for the use of steam flow to determine core thermal power. The analysis indicates that the error due to steam compressibility at 1250 psia is increasingly conservative above 71 percent of full steam flow. (Paragraph 10.f)

3. <u>Review of Plant Operations and Plant Startup After Refueling</u>

The inspector reviewed the following selected operating logs and records, general plant operating conditions, procedures, controls, control room manning, equipment status and tagout status, selected plant annunciators, and selected plant deviation and event reports:

a. Shift Supervisors Log (September 24 through December 1, 1977)

b. Control Room Log (October 1 through December 1, 1977)

c. Reactor Operator Log (October 1 through December 1, 1977)

d. Shift and Auxiliary Logs (November 1 through 30, 1977)

The condensate storage tank level was verified to be above the requirements, but no specific acceptance criteria or checks had been established to insure compliance (This item was discussed at the management exit interview).

e. Daily Order Book (September 9 through December 1, 1977)

f. Plant Master checklist and system checklists including:

(1) O-TGS-1, Rev. 13, 9/23/77 - Master Checklist.

(2) Selected system checklists

Certain of the system checklists needed to be reviewed and revised to reflect up-to-date plant conditions and insure adequate system coverage (This item was discussed at the management exit interview).

(3) Selected instrument checklists

The C-2 and C-3 checklists were reviewed. The checklists for August 26, October 14, and October 22, 1977, contained steps which were not appropriately completed for the startup and intermediate nuclear ranges. The failure to adequately perform the specified checklists is an item of noncompliance pursuant to Technical Specification 6.8.1 and is a deficiency (this item was discussed at the management exit interview).

The inspector verified that the instrument checklists were required to be performed by procedure. A review of the procedural requirements with the appropriate personnel should prevent recurrence. The inspector reviewed the management controls for routing and retaining the completed checklists. The C-2 and C-3 checklists for the May 3, 1976, could not be located. The failure to maintain the checksheets for the startup is an item of noncompliance pursuant to Technical Specification 6.10.1.d and is a deficiency. The licensee management controls established for the retention of records should prevent recurrence of this type problem. No further questions are required of this matter.

g. On October 19, 1977, a short period (10 sec) scram occurred during minimum power operating conditions when control rod E-2 was withdrawn from step 02 to 03 with the period scram in the unblocked, normal status for the plant conditions.

A review of the licensee deviation report (D-BRP-77-143) and the licensee analysis (OSA C15-1, dated October 21, 1977) revealed that the individual step worth of rod E-2 from step 02 to 03 was 0.103 percent reactivity which would result in a stable period of 45 seconds, and this value is well below the Technical Specification maximum noted worth of 0.3 percent reactivity. The trip was apparently caused by the prompt neutron population increase associated with the withdrawal of the control rod.

The licensee returned the plant to power using the alternate rod withdrawal sequence which provided a slight decrease in the noted worth at the specific critical low power conditions and 200°F primary coolant temperature.

The rod withdrawl sequence notch worth for the two approved startup rod sequences indicated a maximum notch worth of approximately 0.165 percent reactivity which is well below the Technical Specification limit of 0.3 percent.

h. At 0130 on November 11, 1977, during control rod drive testing, rod B-4 inserted as required but would not withdraw to the original position. The rod was subsequently scrammed from position 02 to place the rod in the fully inserted position, and simultaneously rods E-3 and C-1 inserted to the full in position from step 06 and 21 respectively. The plant power reduced from approximately 56 MW electrical to 46 MW electrical and the control rods E-3 and C-1 were returned to the normal in-sequence position in accordance with the approved procedure (ONP-2.7.2, Dropped Rod).

A review of the licensee event report (E-BRP-77-58) and the analysis performed by the licensee indicated that the power distribution was apparently satisfactory (conservative by 25 percent) during the reduced power transient. In the close-coupled core an insertion of a control rod (negative reactivity) results in a power reduction and an increase in the safety margin for MCHFR and MAPHLGR limits even though a slight flux tilt may occur.

The licensee issued a temporary procedure change on November 11, 1977, to operate outside the approved control rod program as specified in Volume 5, Section 15.5.A.3 (Rev. 7) of the system operating manual.

The inspector noted that a problem could exist if the rods could not have been returned to the normal sequence and a normal plant power reduction to minimum power or to the shutdown condition was required with the rod(s) out of sequence. A point in the planned power reduction could occur when the alternate sequence would be unacceptable (unanalyzed) and the rod(s) would have to be returned to within the approved sequence or the plant scrammed to prevent operation with an unacceptable control rod pattern with respect to the ejected rod worth requirements (This item was discussed at the management exit). This item will remain open pending further review by the inspector.

j. The inspector reviewed an operational item with the licensee concerning the Technical Specification requirements for performing a control rod drive coupling test prior to each reactor criticality as indicated in sections 7.3.3 and 5.2.2.d.

During the plant startup on October 31, 1977, a question arose concerning a temporary procedure change issued to preclude the coupling integrity test immediately prior to an approach to criticality.

Technical Specification 7.3.3 specifically requires a coupling integrity test to be performed during a plant "Hot Startup" .... "in accordance with section 5.2.2.d. The startup shall then proceed in accordance with paragraphs (d) through (k) of section 7.3.2 ....". Specifically section 5.2.2.d requires "Control Rod Following Verification During Reactor Operation - During each approach to criticality, control rods withdrawn before k_{eff} reaches 0.995 (determined on the basis of predictions or sub-critical multiplication measurements) shall be verified by a coupling integrity check at the time the rod is fully withdrawn. Each control rod which is withdrawn at any time k_{eff} is equal to or greater than 0.995 shall be checked to verify proper rod following before the total worth of the withdrawn portion of the rod and any other unverified rods reach 0.01 Δk (as determined from the approximate Control Rod Worths resulting from the initial calculations and start-up t st data). Verification shall consist of either a coupling intergrity check or the observation of nuclear instrumentation response to rod withdrawal."

The licensee is reviewing the instrument data to assure that control rod coupling was verified during the startup. The practice of routinely performing control rod coupling checks utilizing the nuclear instrumentation on all control rods withdrawn when k_{eff} is equal to or greater than 0.995 is also being evaluated. Individual control rod worth data is available for cycle 15 to establish the worth of unverified rods if the licensee decides to use this method to perform control rod coupling verification with the nuclear instrumentation at or greater than a k_{eff} of 0.995. The licensee routinely performs the coupling checks on all of the rods immediately prior to a critical approach by individually withdrawing one rod at a time to the full out position and verifying no overtravel condition (uncoupled rod).

j. The inspector reviewed an operational event concerning the control rod drive system which occurred on October 31, 1977. During the control rod testing prior to plant startup (TV-07, CRD Scram Test From Notch 03 and Coupling Integrity Checks) in the "jog-bypass" mode, rod drive B-5 moved abnormally rapid from step 02 to step 12 (a few seconds). The operator immediately applied an insert signal and the rod inserted equally as fast from step 12 to step 01. The licensee evaluation of the event is continuing to determine any problem with the control rod drive or the drive system. The rapid withdrawal and insertion did not reoccur during subsequent testing on the rod.

The control rod drives are moved in the single-notch mode during all conditions except shutdown testing; therefore the single event on the one control rod drive during shutdown testing conditions does not appear to be a safety problem during normal plant operations.

The inspector discussed the matter with the licensee representative and reviewed the control rod drive systems, system operating procedures, and management controls insuring the rod control system remains in the single-notch mode except for shutdown testing.

No further questions are required on this matter at this time. The matter will remain open pending completion of the review of the event by the licensee and the inspector.

k. The inspector reviewed selected tests performed during the recent outage and subsequent plant startup. The tests reviewed include:

(1) TR-06, Liquid Poison System Check Valve Test.

A change was made to the pressure instrument calibration sheet concerning the required accuracy of the pressure gauge used during the testing procedure (This item was discussed at the management exit interview).

(2) TR-08, Core Spray System Check Valve Test.

The acceptance criteria for the check valve leakage was established from the prior test and written into the procedure by the licensee representative prior to conducting the test. The check valve leakage was less than the entered acceptance criteria (This item was discussed at the management exit interview.)

(3) TR-17, Instrument and Control Transformer Auto Transfer Test.

(4) TR-29, Containment Vacuum Relief Operability Test, Rev. 4, September 16, 1977.

(5) TR-18, Essential Power Transfer Test.

(6) TR-21, (TV-06), Control Rod Drive Friction Testing Pressure, Rev. 0, March 9, 1976.

(7) TR-01, (T180-08), Control Rod Drive Performance Test, Rev. 1, March 9, 1976.

(8) TR-41, Transmission Line Transfer Test, Rev. 1, July 27, 1977.[1/]

A procedure change was made to step 5.8 to correct the procedure for actual plant conditions during the performance of the test. No temporary procedure change was performed on September 25, 1977. The temporary procedure

1/ IE Inspection Rpt No. 50-155/76-12.

change was issued subsequently on September 29, 1977. The failure to issue the temporary procedure change in accordance with requirements is an item of noncompliance (deficiency) pursuant to Technical Specification 6.8.3 (This item was discussed at the management exit interview). No further questions are required of this matter at this time.

(9) TR-24, Emergency Condenser System Valve Test, Rev. 2, February 17, 1976.

(10) TR-42, Emergency Diesel Generator Full Load Test, Rev. 1, September 19, 1977.[2]

(11) TR-30, Emergency Diesel Generator Protective Device Testing, Rev. 5, July 28, 1977.

The test was performed on August 19, 1977, verifying the fuel oil level, fuel oil pressure, lube oil pressure and trips, water jacket temperatures and trips, overspeed trip, battery undervoltage alarm, and relay timer setpoints. The inspector noted that the cranking timer (TT) setpoint was found at 11 seconds and changed to 13.5 seconds in accordance with the procedure acceptance criteria (13.0 to 13.9 seconds.) The cranking timer (TT) alarm setpoint apparently had not been changed during a previous facility change to change the emergency diesel cranking cycle to 25 seconds continuous cranking (This item was discussed at the management exit).

(12) TR-43 (T180-07/BRP-RE-08), Shutdown Margin Check, Rev. 5, August 2, 1977.

(13) TR-43, Moderator Temperature Coefficient Check.

(14) TR-46 (BRP-RE-07), Fuel Bundle Core Loading Procedure, Rev. 5, August 23, 1977.

(15) TR-55, Fuel Dry Sipping Procedure, Rev. 4, May 13, 1977.

The inspector reviewed a deviation report issued concerning the failure to follow the specified fuel sipping steps resulting in the use of an improper fuel sipping time. The licensee revised the procedure to reduce the chance of error (This item was discussed at the management exit interview).

2/ IE Inspection Rpt No. 50-155/76-12.

(16) TR-63, 1A-2B and 2A-2B, Breaker Shunt Trip Test, Rev. 0, June 15, 1977.[3/]

(17) TR-62(5), Spare Battery Charger Capacity Test, Rev. 0, July 30, 1977.

(18) TR-58, Calibration of Mechanical Vacuum Trip and Alarm, Rev. 0, February 14, 1977.

(19) TV-07, CRD Scram Test from Notch 03 and Coupling Integrity Checks, Rev. 0, July 26, 1977.

(20) TV-10, Hydrostatic Test of the Nuclear Steam Supply System, Rev. 9, September 21, 1977.

l. The inspector reviewed selected plant deviation reports to ascertain adequate and timely licensee corrective actions, reportability, and management reviews.

(Certain deviation reports were discussed at the management exit interview.)

No other items of noncompliance or deviations were identified.

4. Facility Tour

The inspector reviewed general plant and control room conditions during facility tours while onsite. Several items were reviewed including:

a. Status of safety related equipment.

b. Selected plant annunciators.

c. Recent plant changes.

d. Equipment tagout status.

e. Selected logsheet readings and chart recordings.

f. During the tour to the containment through the personnel access hatch, the inspector noted that the alarm bell portion of the personnel access hatch alarm system was inoperative. The red intermittent light appeared to be functioning properly

3/ LER 50-155/77-27.

and the inspector noted that the mechanical interlock/equalizing handle was in the appropriate (down) position maintaining containment integrity (This item was discussed at the management exit interview).

No items of noncompliance or deviations were identified.

5. Licensee Nonroutine Event Reports

The inspector reviewed the licensee actions completed concerning the following nonroutine event reports to verify that the events were reviewed and evaluated as required, the corrective actions were taken, and the plant limits were not exceeded. The review included selected records, meeting minutes, and interviews with plant personnel.

a. LER 77-25 and 77-28, Automatic Containment Isolation valve (CU-4093) Failure to Close Properly.[4/] [5/]

b. LER 77-32, Primary Core Spray Deficiency.[6/]

The primary ring core spray header has been reviewed by NRR and plant operation has been permitted under an examption request as allowed by Amendment 15 to the Technical Specifications.

c. LER 77-36, Control Rod Withdrawal Times in Excess of Requirements.[7/]

d. LER 77-37, Diesel Fire Pump Failure to Start.[8/]

e. LER 77-38, Feedwater Check Valve Excessive Leakage.[9/]

The initial Type A Containment test revealed excessive leakage of the feedwater check valve. The feedwater line was isolated and the Type A test completed satisfactorily. The feedwater check valve was subsequently repaired and successfully leak tested on September 24, 1977. The licensee will submit the test data for review to the NRC pursuant to Section V.B. of Appendix J to 10 CFR 50.

f. LER 77-39, Reactor Depressurization Battery "A", Cell 12, Low Specific Gravity.[10/]

g. LER 77-40, Core Spray Initiation Pressure Sensors Below the Technical Specification Requirements.[11/]

4/ Ltr, CP to RIII, dtd 9/16/77.
5/ Ltr, CP to RIII, dtd 8/25/77.
6/ Ltr, CP to RIII, dtd 9/23/77.
7/ Ltr, CP to RIII, dtd 9/23/77.
8/ Ltr, CP to RIII, dtd 9/29/77.
9/ Ltr, CP to RIII, dtd 9/29/77.
10/ Ltr, CP to RIII, dtd 10/27/77.
11/ Ltr, CP to RIII, dtd 11/1/77.

The corrective actions to prevent recurrence were reviewed by the inspector with the licensee representative (This matter was discussed at the management exit interview.)

No items of noncompliance or deviations were identified.

6. Records

The inspector reviewed selected plant records to ascertain that control, storage, retention, and retrieval of records and documents were in accordance with the requirements. The inspector noted that the licensee was continuing to complete the implementation of the record storage system at the plant. The records reviewed by the inspector were available and retrievable.

No items of noncompliance or deviations were identified.

7. IE Bulletins and Circulars

The inspector reviewed the licensee's actions taken concerning selected IE Bulletins and Circulars.

a. IEB 77-02, Potential Failure Mechanism in Certain Westinghouse AR Relays with Latch Attachments - September 13, 1977.[12]

b. IEC 77-09, Improper Fuse Coordination in BWR Standby Liquid Control System Control Circuits - May 27, 1977.

The control fuses at the facility appear adequate and are routinely tested. The monitoring circuit was recently modified by the licensee to provide improved alarming of a circuit failure.

c. IEC 77-10, Vacuum Conditions Resulting in Damage to Liquid Process Tanks - July 19, 1977.

d. IEC 77-11, Leakage of Containment Isolation Valves with Resilient Seats - September 7, 1977.

e. IEC 77-12, Dropped Fuel Assemblies at BWR Facilities - September 16, 1977.

f. IEC 77-13, Reactor Safety Signals Negated During Testing - September 23, 1977.

12/ Ltr, CPC to RIII, dtd 10/26/77.

The review of selected testing procedures during plant power operations and outages, and the maintenance controls established revealed no apparent problems relative to bypassing or negating plant protection signals.

No items of noncompliance or deviations were identified.

8. Amendment 15 to the Technical Specifications[13/]

The inspector reviewed selected items covered by the amendment to the plant technical specifications and the written safety analysis for the amendment. The areas reviewed included:

a. Emergency Procedures (EMP)

   (1) Training and walk through for operations personnel.

   (2) Main feedwater and condensate systems operations during a LOCA.

   (3) Procedure deletions concerning selective isolation of a core spray header.

   (4) Curtailment of firefighting activities until the long term core cooling phase of the LOCA is established.

b. Off-normal Procedures (ONP)

   (1) Loss of Feedwater System

   (2) Loss of Condensate System

c. Surveillance Procedures

   (1) Monthly verification of the spray header crossconnect valve in the open position and sealed locally.

   (2) Monthly verification of the core spray heat exchanger backup hoses on the hose cart and available.

   (3) Monthly verification of the fire protection system deluge valve closed and the manual isolation valve closed and sealed. The inspector noted a discrepancy in the procedure during the verification of the valves in the required positions (This item was discussed at the management exit interview).

13/ Ltr, NRR to CPC, dtd 10/17/77.

(4) Monthly verification of the condenser hotwell fill valve (CV-4009) operability and the manual isolation valve (C-32) open and sealed (This item was discussed at the management exit interview).

(5) Monthly operability checks of the core spray system valves (MO-7051, MO-7061, MO-7070, MO-7071, and MO-7066).

d. Standard Operating Procedures

(1) Fire protection system supply to the hotwell manual isolation valve closed and locked (SOP-26).

(2) Two condensate pumps required for operation above 50% reactor power (SOP-15).

(3) Condensate storage tank level at least 65% full during power operations (This item was discussed at the management exit).

(4) The backup containment spray valve (MO-7068) closed with the electrical breaker open and locked (SOP-8).

(5) No fire protection system hydrant flushing or fire pump capacity tests to be run during plant power operations (SOP-26).

(6) Operation of the condensate and feedwater systems during the LOCA (SOP 15 and 16).

(7) Use of the fire protection supply to the core spray heat exchanger and the back core spray valve MO-7072.

(8) Use of the LOCA-qualified reactor pressure indicator (SOP-8).

(9) Use of the core spray flow recorder and test circuitry (SOP-8).

9. Quality Assurance

The inspector reviewed the completed and scheduled plant quality assurance audits for 1977. Selected audit findings were reviewed to assure adequate and timely corrective actions. Changes to the quality assurance program were reviewed.

The licensee scheduled thirteen audits for 1977 and has completed eight audits.

No items of noncompliance or deviations were identified.

10. Safety Limits, Limiting Safety System Setpoints, and Limiting Conditions for Operations

The inspector reviewed selected licensee records and procedures including charts, recordings, surveillance test procedures, calibration procedures, maintenance procedures, deviation reports, event reports and operating logs to insure that facility was operated within the requirements.

No items of noncompliance or deviations were identified.

11. Outstanding Inspection Items

The inspector reviewed selected outstanding inspection items to determine appropriate actions completed by the licensee.

a. Nuclear Instrument Cable Failures[14/]

The licensee has replaced the nuclear instrument channel cables with high temperature cables as the channel cables have deteriorated. The new cables have demonstrated satisfactory operation. The startup channel cables remain to be replaced. No further questions are required of this matter at this time and this matter is considered closed.

b. Emergency Diesel Generator Redundant Trip Circuitry[15/]

The licensee has completed modifications to the emergency diesel generator to provide redundant tripping from high water jacket temperature, low lube oil pressure, and overload protection. No further questions are required of this matter at this time and this item is considered closed.

c. LOCA Qualified Pressure Transmitter[16/][17/]

The licensee completed the installation and checkout of a LOCA qualified pressure channel which can be used during the loss of coolant accident to monitor reactor coolant pressure. The

14/ IE Inspection Rpt No. 50-155/75-12.
15/ Ltr, CPC to NRR, dtd 7/26/77.
16/ Ltr, CPC to NRR, dtd 6/2/76.
17/ IE Inspection Rpt No. 50-155/77-12.

inspector noted an apparent discrepancy in the design relating to seismic requirements (This matter was discussed at the management exit). No further questions are required of this matter at this time and this item is considered closed.

d. Air Ejector Offgas Monitor Failure[18/][19/]

The licensee completed a modification to the offgas system which provides insulation and heat tracing to selected portions of the system to prevent moisture buildup during plant operation. No further questions are required of this matter at this time and this item is considered closed.

e. Single Loop MAPLHGR Limits[20/]

The operating requirements were reviewed by NRR and the plant limits issued with amendment 15 to the Technical Specifications.[21/] No further questions are required of this matter at this time and this item is considered closed.

f. Steam Flow for Heat Balance Calculations[22/]

The licensee submitted an error analysis[23/] for the use of steam flow to determine core thermal power. The error due to steam compressability at 1250 psia becomes increasingly conservative above 71 percent full steam flow and therefore provides an accurate parameter to measure total power. No further questions are required of this matter at this time and this item is considered closed.

g. Movement of the Dunker Type Neutron Detectors During Refueling[24/]

The dunker type startup detectors (channels 8 and 9) were used during the fuel loading procedure (RE-02) and the shutdown margin testing procedure (TR-43). No further questions are required of this matter at this time and this matter is considered closed.

18/ LER 50-155/77-13.
19/ IE Inspection Rpt No. 50-155/77-07.
20/ LER 50-155/77-20.
21/ Ltr, NRR to CPC, dtd 10/17/77.
22/ IE Inspection Rpt No. 50-155/77-08.
23/ Ltr, CPC to RIII, dtd 8/10/77.
24/ IE Inspection Rpt No. 50-155/77-08.

h. Nonconservative MCHFR Limits Inputs to the Online Computer[25/]

The licensee is no longer utilizing the online computer to indicate MCHFR limits due to the inadequacy of the program. All MCHFR limits are established manually and provided to the operations group. The inspector noted the operations group instructions were given for operating at 205 MW_t which provided ample margin to MCHFR limits. No futher questions are required of this matter at this time and this item is considered closed.

i. Visual Inspection of Containment Vessel Mechanical Penetrations and Sleeves[26/][27/]

The licensee visually inspected all accessible containment mechanical penetrations and sleeves (TV-01, Rev. 1, December 10, 1976). No further questions are required of this matter at this time and this item is considered closed.

j. Control Rod Drive Pump Suction Line Check Valve[28/]

The licensee installed and tested a suction line check valve for the CRD pumps to provide containment integrity requirements of the CRD pump suction line in conjunction with the individual pump internal poppet valves.

k. Control Rod Drive Pump Cooling Water Line Check Valves[29/]

The licensee has installed and tested the check valves in the CRD pump piston cooling water lines to provide containment integrity requirements.

l. Containment Isolation Valve (CV-4093)[30/][31/]

The licensee has cleared a mechanical binding on the isolation valve which was preventing the valve from fully closing and other similiar valves were inspected to insure no mechanical binding existed. The licensee modified the standard operating procedure (SOP-3) to insure flushing of resins from the valves following use to reduce the chance of internal binding. No futher questions are required of this matter at this time and this item is considered closed.

25/ IE Inspection Rpt No. 50-155/77-08.
26/ LER 50-155/76-27.
27/ IE Inspection Rpt No. 50-155/76-21.
28/ LER 50-155/77-29.
29/ LER 50-155/77-30.
30/ LER 50-155/77-25.
31/ LER 50-155/77-28.

m. Containment Vacuum Relief System[32/33/34/35]

The licensee completed the modifications and testing of the containment ventilation valves to provide an improved vacuum relief system for the containment building. The modification includes the use of redundant vacuum and pressure channels and a vacuum relief flow path via the ventilation supply or exhaust valves. The modification also provides a channel failure interlock to prevent defeating of the containment isolation function due to a channel failing downscale. No futher questions are required of this matter at this time and this item is considered closed.

n. Reactor Depressurization System Circuitry Testing at Shutdown[36/37]

The licensee completed operational testing of the reactor depressurization system to verify the complete system operability by manually testing of the individual channel, channel logic, and valve operations. The testing provides increased assurance that the automatic test clock feature has functioned adequately. No futher questions are required of this matter at this time and this item is considered closed.

o. Automatic Containment Isolation[38]

The licensee completed the modification to the containment ventilation valves to provide automatic closure from high radiation levels at the new or spent fuel storage areas. No further questions are required of this matter at this time and this item is considered closed.

p. Reactor Depressurization System Valve Testing at Shutdown[39/40]

The reactor depressurization valves (Target Rock) were pressurized and tested in accordance with procedure SST-02, RDS Valve Testing, on August 4, 1977. No further questions are required of this matter at this time and this item is considered closed.

32/ IE Inspection Rpt No. 50-155/75-05.
33/ IE Inspection Rpt No. 50-155/76-01.
34/ IE Inspection Rpt No. 50-155/76-12.
35/ IE Inspection Rpt No. 50-155/76-18.
36/ LER 50-155/76-42.
37/ IE Inspection Rpt No. 50-155/77-01.
38/ Ltr, NRR to CPC, dtd 2/6/77.
39/ LER 50-155/76-42.
40/ IE Inspection Rpt No. 50-155/77-01.

q. Reactor Depressurization System Isolation Valve Position Switches[41/42/]

The licensee completed the modification to the reactor depressurization system isolation valve position switches which placed the switch in the neutral leg of the circuit to prevent a failure to ground and tripping of the valve power supply. New qualified switches are being procured which will be installed upon arrival onsite. No further question are required of this matter at this time and this item is considered closed.

r. Fuel Heatup Analysis Nonconservative Limits[43/]

The fuel limits for the Cycle 15 fuel were reviewed by NRR and the plant limits issued with amendment 15 to the Technical Specifications.[44/] No futher questions are required of this matter at this time and this item is considered closed.

s. Station 46 KV Transformer No. 7 Bushing Failure[45/46/]

The bushing faulure was apparent random component deterioration which was discovered during a routine test. No further questions are required of this matter at this time and this item is considered closed.

12. Management Exit Interview

The inspector conducted a management interview with the licensee representatives (denoted in paragraph 1) at the plant on November 18, and December 2, 1977. The inspector summarized the scope and findings of the inspection. The licensee made the following remarks concerning certain items discussed by the inspector.

a. The licensee stated that the system valve checklists would be reviewed and updated as appropriate. (Paragraph 3.f)

b. The licensee stated that the acceptance criteria for the condensate storage tank level would be reviewed and the level would be checked routinely by the operations management rather than change the log for one operating cycle only. (Paragraph 3.d and 8.d)

c. The licensee stated that the inoperative alarm bell system on the containment personnel access hatches was only partially operable (lights) while new bells were being procured. The alarm lights and the posted procedures were being used to insure containment integrity was being maintained. (Paragraph 4.f)

41/ LER 50-155/77-04.
42/ IE Inspection Rpt No. 50-155/77-02.
43/ LER 50-155/76-29.
44/ Ltr, NRR to CPC, dtd 10/17/77.
45/ LER 50-155/77-22.
46/ IE Inspection Rpt No. 50-155/77-12.

d. The licensee stated that the failure to properly complete the startup checksheets (C-2) on August 26, October 14, and October 22, 1977, was a personnel error and the sheets would be routed for review and corrective action. The checksheets for startup on May 3, 1976, could not be located. (Paragraph 3.f)

e. The licensee stated that the condensate fill valve was flow tested prior to plant startup. (Paragraph 8.c)

f. The licensee stated that the surveillance procedure concerning amendment 15 items would be corrected to indicate the fire protection system deluge isolation valve closed and sealed and that the inadequacy in the procedure was deemed reportable because of the potential for degrading the ECCS requirements if the procedure had been performed as indicated. (Paragraph 8.c)

g. The licensee commented that the operation of the plant with a rod inserted resulting in an unacceptable rod pattern at minimum power or hot shutdown, no voiding condition would be reviewed. (Paragraph 3.h)

h. The licensee stated that the voltage study relative to the Millstone Event[47/] was in progress and the company was continuing the testing and evaluation, including the station transformer regulator being required to maintain the station power supply voltage to within 5 percent of nominal.

i. The licensee stated that the review of the plant connectors revealed no apparent problems relative to the LOCA; since the connectors located in the plant systems are in the rod position indication and the liquid poison system, which are not directly related to the ECCS requirements.

j. The licensee stated that the failures of the emergency diesel generator to start adequately on October 20 and November 24, 1977, were under review and fuel oil samples had been taken and the use of a premium grade fuel oil was being evaluated. (Paragraph 5).

k. The licensee indicated that the deviation reports concerning the failure to follow procedures, exceeding the administrative limit on thermal power, and the severence of the control cable to the well house would be reviewed. The licensee stated that offsite corporate licensing assistance has been requested concerning the failure to follow procedures being a technical specification violation requiring subsequent PRC review as indicated by the inspector pursuant to Technical Specification, section 10/6.5.1.6.e. (Paragraph 31)

47/ Ltr, CPC to NRR, dtd 12/7/76.

l. The licensee stated that the management controls to prevent recurrence of items such as the failure to change the setpoints on the core spray pressure channels following a Technical Specification change would be reviewed. (Paragraph 5.g)

m. The licensee stated that the facility change package on the LOCA-qualified reactor pressure channel would be reviewed in regards to seismic requirements, and the use of channel checks would be considered. (Paragraph 11.c)

n. The licensee stated that the minor procedural discrepancies in the various refueling surveillance and test procedures would be reviewed including: no temporary procedure changes issued, failure to follow procedures, the apparent failure of supervision to note the procedural discrepancies during routine review, and initiate the appropriate corrective action. (Paragraph 3.k)