U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-155
License No:	DPR-06
Report No:	50-155/96006(DRP)
Licensee:	Consumers Power Company
Facility:	Big Rock Point Nuclear Power Plant
Location:	10269 U.S. 31 North Charlevoix, MI 49720
Dates:	July 27 - September 06, 1996
Inspectors:	R. J. Leemon, Senior Resident Inspector C. E. Brown, Resident Inspector
Approved by:	W. J. Kropp, Chief, Projects Branch 3 Division of Reactor Projects

EXECUTIVE SUMMARY

Big Rock Nuclear Power Plant
NRC Inspection Report 50-155/96006

This routine 6-week inspection covered aspects of licensee operations, engineering, maintenance, and plant support. Additionally, this report includes the results of announced inspections by regional specialists.

Operations

- Operating crews performed well during on-the-job training and off-normal situation communication. However, there was a weakness in the quality of turnovers by ROs leaving the control room for a short period of time (Section O1.2).

- An operating crew declared check valve, VPI-303, in the core spray backup system operable even though the valve did not meet acceptance criteria during a surveillance test. Declaring the valve operable is unresolved item pending further review of the operability determination process by the NRC.

Maintenance

- The documented operability determination for excessive leakage past check valve, VPI-303, focused only on VPI-303 and did not fully document the basis for the backup core spray system operability as affected by degraded VPI-303 operation. (Section M3.1).

Engineering

- The inspectors were concerned that the post incident system relief valves could stick open during an actual demand which would result in decrease flow in the reactor (Section E1.1). The licensee has a plant life time exemption for a LOCA in one core spray train and a single failure in the remaining operable core spray train which could be a stuck open relief valve. Adequate core spray flow with a stuck open PI relief valve with a line break in the one of the core spray trains is considered an inspection follow-up item pending further NRC review (50-155/96006-02(DRP)).

- The investigation and analysis of the failed 86R pilot valve were technically sound, and the NRC staff had no concerns with the licensee's operability determination for the four installed RDS-pilot valves (Section E2.1).

Plant Support

- The work activities to radiation profile of the spent fuel channels and to clean the spent resin tank room were well supervised and completed as planned without exceeding the dose estimate (Section R1.1).

Report Details

Summary of Plant Status

The plant operated at full power throughout the period, and no events occurred which challenged the control-room operators.

I. Operations

01 Conduct of Operations

01.1 General Comments

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety-conscious; specific events and findings are detailed in the sections below.

01.2 Shift Manning and Formality

a. Scope (71707)

The inspectors observed the conduct of shift supervisors, nuclear control operators, and auxiliary operators during shift turnovers, plant rounds, and surveillance testing. These observations were compared to the plant technical specification (TS) and final hazards summary report (FHSR) requirements.

b. Observations and Findings

The minimum required shift manning, including two reactor operators (ROs), was maintained. However, during several shift turnover periods, one of the on-shift ROs would leave the control room for a short period of time without ensuring that the other RO was fully cognizant of the plant status. The turnover consisted only of the statement, "Stepping out," and contained no information related to plant equipment or status. This statement was also routine for periods between shift turnovers. Pending further review by the licensee and the NRC, the adequacy of shift turnovers is considered an unresolved item (50-155/96006-01(DRP)).

Auxiliary operator (AO) performance during a diesel fire pump surveillance test was good. The AO asked personnel to clear the area in-front of the diesel before starting the test in order to take readings immediately after the diesel started. The AO also frequently checked the condition and parameters of the diesel during the 30 minute diesel run.

During turnovers, the inspectors observed that the AOs-in-training demonstrated the same good traits and acquired skills as the experienced AOs. The experienced AOs provided good on-the-job-training to these AOs in the following areas:

- Pre-job briefings
- Walk-throughs, including tour techniques, both inside and outside containment
- Equipment tagouts for maintenance
- Communication techniques (repeat backs).

Newly licensed ROs were conscientious about the plant status and used good communication techniques when talking to either the other RO or to the AOs in the plant.

The shift supervisor also had good communication with management during the failure of core spray check valve VPI-303 to seat (Section M3.1). Shift supervisor's communication with management during off-normal situations had been noted as a weakness by the resident inspectors in IR 50-155/96002, Section E2.1, relating to a turbine trip. These recent communications demonstrated an improvement in comparison to the turbine trip event.

c. Conclusion

Operating crews performed well during on-the-job training and off-normal situation communication. However, there was a weakness in the quality of turnovers by ROs leaving the control room for a short period of time.

O8 Miscellaneous Operations Issues (92700)

O8.1 (Closed) Violation 50-155/95006-07: Wide-Range Monitors (WRMs) High Neutron Flux Setting Exceeded TS Allowable Limits. Two of three neutron WRMs were allowed to become out-of-calibration low during power escalation following a forced outage. The high power reactor scram would not have occurred until 132 percent of rated core power vice the TS allowed 120 $\pm$ 5 percent. This occurred because the WRMs were recalibrated at 63 percent power during the power escalation and the follow-on crews didn't understand the potential for the WRM readings to diverge at higher power levels when the control rods were pulled further out of the core. Additionally, the operators failed to ensure that each WRM channel indicated $\geq$ actual core power during the remainder of the escalation to full power.

The inspectors have monitored several power reductions or escalations since this occurrence in early 1995. In each instance, the operators were very attentive to the WRM calibration. Additionally, the WRMs indications were frequently verified by other control room indications of core power. The training given to the operating crews was effective

as verified by individual interviews with multiple operators. Shift turnovers during power changes have been controlled and formal. The inspectors concluded that the licensee's corrective actions had been appropriate. This item is closed.

O8.2 <u>(Closed) Violation 50-155/95006-01c:</u> See event description in Section O8.1. This item is closed.

O8.3 <u>(Closed) Violation 50-155/95006-04b:</u> See event description in Section O8.1. This item is closed.

O8.4 <u>(Closed) Violation 50-155/95006-08:</u> See event description in Section O8.1. This item is closed.

II. Maintenance

M1.1 Conduct of Maintenance

M1.1 <u>General Comments</u>

a. <u>Inspection Scope (62703) (61726)</u>

The inspectors observed all or portions of the following work activities:

<u>Maintenance Activities</u>

- WO MSS 12611970 replace "B" turbine bypass hydraulic oil pump unloader
- WO SPS 12512030 test spare breaker
- WO SPS 12511884 functional inspection circuit breaker 052-1E21
- WO EPS 12610230 monthly battery readings (battery-2-C)
- WO PIS 12611813 rebuild RV-5082
- WO FPS 12611812 provide seal for fire barrier
- WO RDS 12610160 disassemble, inspect, repair, test SV-4987 pilot valve
- WO RDS 12610175 pilot valve leaks (S/N 86R-4)
- WO RDS 12611864 adjust CV-4183 packing
- WO SPS 12610527 test breaker and contractor
- WO ASD 12610198 monthly battery readings (ASD)
- WO BLS 12611162 loading dock has deteriorating concrete
- WO EPS 12610186 monthly battery readings (Battery-1)
- WO BLS 12611334 repair fuel cask loading dock concrete piers
- WO FPS 12611891 adjust engine speed switch
- WO RDS 12412099 RDS pilot valve 73V-14 leaks through
- WO RWS 12611061 test sump pump for resin tank room repairs

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Surveillance Activities

- T1-02: primary system leakage test
- T1-09: heat balance calculation
- T7-04: reactor protection system logic test
- T7-20: diesel fire pump auto start
- T7-28: emergency diesel generator (EDG) auto test start
- T7-33: check of ASD equipment
- T30-22: ECCS valve test
- T30-44: containment Gamma monitor monthly channel check
- T90-07: RDS isolation valve test operate at power
- T90-25: partial stroke test of MSIV (MO-7050)
- TR-65A: RDS "A" battery service test and discharging alarm operability verification
- TR-65C: RDS-UPS "C" battery service test and discharging alarm operability verification
- TR-65D: RDS-UPS "D" battery service test and discharge alarm operability verification
- TV-16C: RDS-UPS "C" battery discharge test
- TV-16D: RDS-UPS "D" battery discharge test

b. Observations and Findings

The mechanics and operators used the procedures at the job site, and job status was frequently monitored by visits from supervisors and system engineers. On-the-job training (Section 01.2) and assistance to engineering (Section E2.1) were particularly noteworthy. Where applicable, proper radiation control measures were in use.

c. Conclusion

The inspectors found the work done under these activities to be correctly performed and accurately documented.

M3 Maintenance Procedures and Documentation

M3.1 Inadequate Safety Evaluation for Leakage through Tell-tale During Testing of Backup Core Spray Check Valve (VPI-303)

a. Inspection Scope

On August 27, 1996, at 8:33 p.m., the licensee while performing surveillance test T30-22, emergency core cooling system (ECCS) valve test, observed an approximate 4-foot plume of steam coming out of a tell-tale. This test included stroke timing the motor operated primary and backup (BU) core spray valves. The steam plume indicated that a check valve, VPI-303, was not fully seated.

The inspectors monitored the leakage through the tell-tale, attended licensee discussions and PRC meetings, reviewed surveillance test data, and observed the gathering of test data related to valve position and tell-tale pressure.

b. Observations and Findings

Test procedure T30-22 required the AO to verify that check valve VPI-303 was adequately seated by checking flow through the tell-tale between MO-7070 and MO-7071 while MO-7071 was open. The AO documented that valve VPI-303 failed the acceptance criteria, "Less than a steady stream from the tell-tale," by marking the "NO" box in the procedure and notifying the shift supervisor (SS). A second attempt to seat VPI-303 by re-stroking MO-7071 was unsuccessful.

The AO notified the shift supervisor (SS) of VPI-303's failure to seat after MO-7071 was stroked open per T30-22. A second attempt to seat VPI-303 by re-stroking MO-7071 was also unsuccessful. At 9:07 p.m., the SS informed the on-call manager and the NRC resident inspector that the second attempt to seat VPI-303 had failed. The SS and on-call manager held discussions and concluded that the backup core spray would perform its intended safety function. The SS then held a discussion with the operation's manager who concurred with the conclusion and to the directions the SS had given his crew to increase monitoring the temperatures between MO-7070 and MO-7071 -- to ensure that the temperature remained less than 280 °F (about 50 psig). The SS did not declare VPI-303 inoperable even though the acceptance criteria was not met. The SS did initiate a condition report to document the problem with VPI-303. The ability to declare a component operable even though the component did not comply with the acceptance criteria defined in the surveillance procedure is considered an unresolved item pending further NRC review (50-155/96006-02(DRP)).

On August 28, the plant review committee (PRC) reviewed the event and determined that the valve VPI-303 and the backup core spray system should have been declared inoperable as of the time that the AO noticed the excessive leakage past the tell-tale. A 24 hour LCO was then entered. A team formed to evaluate the VPI-303 leakage proposed a change to the acceptance criteria and a revision to T30-22 to install a pressure transducer to record tell-tale pressure. These proposed changes were approved by the PRC, and based on tell-tale pressure readings observed while developing the acceptance criteria change, valve VPI-303 and the backup core spray system were declared operable. Additionally, the PRC reviewed and approved the associated safety evaluation.

The resident inspectors and NRC staff reviewed the safety evaluation and concluded that the scope was too focused on VPI-303 allowable leakage. The licensee did not initially provide the pressure rating of the piping, the type of valves, assurance that MO-7070 and MO-7071 would open with a plugged tell-tale, nor what routine monitoring was performed to ensure that the tell-tale was not plugged. If the backup core spray system was required to operate to provide cooling to the reactor core, water would flow from the fire system through the backup core spray system, controlled by motor-operated gate valves MO-7070 and MO-7071.

The NRC staff asked the licensee if valves MO-7071 and MO-7070 could operate under design basis accident conditions if the spool piece between the two valves was pressurized to a pressure of 1335 psig. The licensee determined that the high pressure piping included both the spool piece and MO-7070. Additionally, the licensee concluded that the spool piece would not pressure lock at 1335 psig for the following reasons:

- Valve MO-7071 was tested monthly with 1335 psid across the disk and would open against a differential pressure of 1335 psi with a safety margin of 13 per cent using a calculation that included degraded voltage. Additionally, this valve, a flexi-wedge-gate valve, was installed with a machined notch on the reactor side disc to prevent pressure binding between the discs and allow any high pressure trapped in the spool piece to relieve to the reactor.

- After MO-7071 opens, any residual pressure would be relieved and MO-7070 would be able to open.

- Due to blockage, the tell-tale was modified in 1991 by moving the hole from the bottom of a pipe plug to a drilled 1/8-inch hole in the side of the piping, a location less likely to plug. In accordance with procedure T30-22, the hole is checked clean monthly. Additionally, the pipe plug is removed from the tell-tale and any accumulated corrosion products are removed and sent to engineering for analysis.

c. Conclusion

The documented operability determination for excessive leakage past VPI-303 focused only on VPI-303 and did not fully document the basis for the backup core spray system operability as affected by degraded VPI-303 operation. As a result, the NRC staff had extensive communication with the licensee to fully understand the basis for operability in regards to the backup core spray system. The licensee subsequently documented the operability evaluation to include the backup core spray system.

III. Engineering

E1 Conduct of Engineering

E1.1 Post Incident (PI) System Leaking and Lifting Relief Valves

a. Inspection Scope

The inspectors reviewed the recurrence of the post incident (PI) system relief valves lifting during surveillance testing for applicability to the licensing and design basis.

b. Observations and Findings

The initial pressure surge during diesel fire pump surveillance runs has usually caused one or more of the three PI relief valves, set at 155 psig and supplied from the fire system, to lift. Several of the plant staff expressed acceptance of this response even though it was not expected. A condition report (CR) written for a PI relief valve lifting and not reseating was initially rated as a level 4, trend only. However, the corrective action review board (CARB) reviewed the CR the following day and changed the classification to a level 3 which requires additional evaluation.

The three PI relief valves and associated isolation valves were added during several modifications, and two of the relief valves were required to be in service. During a previous DFP test, one PI relief valve was isolated and caution tagged after leaking. On a subsequent DFP test, another PI relief valve lifted and leaked greater than the acceptance allowance. The licensee cleared the caution tag from the first relief valve and returned it to service, and the second relief valve was isolated. The first relief valve was no longer leaking. The licensee generated a condition report and work requests to repair both of the valves; however, only one of the relief valves was repaired by the licensee during the next few days.

The licensee has accepted the PI relief valves lifting and not fully reseating for many years. The inspectors were concerned that the relief valves could stick open during an actual demand which would result in decrease flow to the reactor. However, the licensee has a plant life time exemption for a LOCA in one core spray train and a single failure in the remaining operable core spray train which could be a stuck open relief valve. The licensee also had a engineering analysis that bounded a stuck open relief valve but was not part of the design basis of the plant. Adequate core spray flow with a stuck open PI relief valve with a line break in the one of the core spray trains is considered an inspection follow-up item pending further NRC review (50-155/96006-03(DRP)).

E1.2 RDS Pilot Valve Internal Sleeve Corroded in Place

a. Inspection Scope

A spare reactor depressurization system (RDS) relief pilot valve failed to function during a test. Upon inspection of the inside of the 86R-4 pilot valve during disassembly, a cracked "tooth" was observed on the sleeve and corrosion was evident on the main disc and sleeve area. The inspectors observed and assessed the licensee's actions in response to this failed pilot valve.

b. Observations and Findings

On August 9, dimensions on spare RDS pilot valve 86R-4 were obtained in preparation for future modifications. The valve had been taken off the

"B" train RDS header during the refueling outage in January 1996, successfully tested, and stored in the turbine building with the openings taped. Upon disassembly, the system engineer determined that the valve sleeve had rusted to the valve body and that corrosion was evident on the valve disc. This immediately raised concerns about the condition and operability of the model 86R pilot valves installed in the RDS system. The licensee had four pilot valves of this model; two were spares and two were installed on the RDS header. On August 12, after discussion with the manufacturer, valve 86R-4 was sent to a laboratory for an inspection which revealed that mainly iron corrosion products were present on the valve disc, the sleeve, and the seating area.

The spare RDS pilot valve 86R-1 was satisfactorily pop tested, disassembled, and inspected; no corrosion was found. The licensee sent the sleeve from the 86R-1 valve to the corporate lab for analysis. The analysis revealed that the corrosion in the 86R pilot valve contained excessive sulphur which was most likely from foreign material in the valve. Also, the spare model 73V RDS pilot valves were disassembled and inspected for comparison to model 86R pilot valves. No corrosion or discrepancies were found.

Based on the laboratory analysis results and the examination of the second spare 86R pilot valve, the licensee concluded that the two installed 86R pilot valves were operable.

c. Conclusion

The investigation and analysis of the failed 86R pilot valve were technically sound, and the NRC staff had no concerns with the licensee's operability determination for the four installed RDS-pilot valves.

E2 Engineering Support of Facilities and Equipment

E2.1 Review of FHSR Commitments

The inspectors compared plant operation and surveillance activities with the Final Hazards Safety Report (FHSR). No deviations were noted during this report period.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radiation Profiling of Used Fuel Channels and Resin-Tank Room Cleaning

a. Inspection Scope

The inspectors observed the licensee preparation and performance of the first phase of the project to clean residual spilled resin from the room and to construct a new piping pathway to transfer spent resin to the resin dump and concentrator tanks. Additional radiological work

consisted of the underwater survey and dose characterization of channel racks and a representative sampling of used fuel channels for isotopic content to determine waste classification and packaging requirements.

b. Observations and Findings

In order to conserve dose, the licensee had hired an experienced contractor to clean the spent resin room. Additionally, the licensee used radio head-sets for communications, tele-dosimetry, and television monitors to remotely monitor dose and job performance. This was the first time tele-dosimetry had been utilized for direct dose surveillance at this site. The total dose was 15.4 person-rem versus an estimated 17.788 person-rem for the project. An interview with the project coordinator revealed that unexpected problems (rebuilding the transfer pump four times due to loss of suction and clearing a plugged spent-resin tank room to radwaste sump drain line) cost 5.3 person-rem.

The cleaning was completed with no spills and reduced the stored radioactive waste on-site by 5-cubic feet. Also, the dose rate in the room was reduced by 200 mrem/hour.

The channel profile work went well even with the use of contractor workers. Controlling dose was a major focus point, and the work was completed under the dose goal for this phase of the project. The licensee had built a temporary shield wall on the refueling deck to reduce the exposure when the supervisor and worker were not on the spent fuel pool (SFP) bridge. An interview with the ALARA coordinator revealed that the wall created a very low dose waiting area, reducing the dose rate from 1.5 to 0.6 millirem per hour (15 to 6 micro-sieverts per hour).

Channel radiation profiles on 88 of 150 channels, now in the SFP, were completed. Dose rates ranged from 10 rem to 13,000 rem on the transition pieces and from 100 rem to 400 rem on the square portions of the channels. The round portions of the channels averaged 2 rem. The licensee concluded that channels from the last core cycle may require some decay time prior to shipment.

c. Conclusion

The work activities to radiation profile of the spent fuel channels and to clean the spent resin tank room were well supervised and completed as planned without exceeding the dose estimate. However, several weak health physics practices identified during the work are discussed below. (Sections R1.2 and R4.1)

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on September 11, 1996. The licensee acknowledged the findings presented.

The licensee did not identify any of the documents or processes reviewed by the inspectors as proprietary.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

- P. Donnelly, Plant Manager
- R. Addy, Assistant Plant Manager
- S. Beachum, Systems and Project Engineering Manager
- E. Bogue, Chemistry/Health Physics Manager
- G. Boss, Operations Manager
- D. Hice, Maintenance Manager
- G. Withrow, Plant Safety and Licensing Director

INSPECTION PROCEDURES USED

IP 37551:	Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observations
IP 62703:	Maintenance Observation
IP 64704:	Fire Protection Program
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 73753:	Inservice Inspection
IP 83729:	Occupational Exposure During Extended Outages
IP 83750:	Occupational Exposure
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92902:	Followup - Engineering
IP 92903:	Followup - Maintenance

ITEMS OPENED and CLOSED

Opened

155/96006-01	URI	Weak Turnovers During Shifts
155/96006-02	URI	Declaring component operable that does not meet acceptance criteria
155/96006-03	IFI	Sufficient core spray flow

Closed

155/95006-07:	VIO	WRM high neutron flux scram setting exceeded TS allowable
155/95006-01c:	VIO	inadequate communications within the shift concerning WRM recalibration
155/95006-08:	VIO	inadequate communication between shifts concerning WRM recalibration
155/95006-04b:	VIO	Inadequate training on WRM operations during power escalation

LIST OF ACRONYMS USED

ALARA	As Low As Reasonably Achievable
AO	Auxiliary Operator
CFR	Code of Federal Regulations
ESF	Engineered Safety Feature
FHSR	Final Hazards Summary Report
HP	Health Physics
IFI	Inspection Followup Item
IP	Inspection Procedure
IPE	Individual Plant Evaluation
IPTE	Infrequently Performed Test and Evolution
IR	Inspection Report
LCO	Limiting Condition for Operation
LER	Licensee Event Report
NCV	Non-Cited Violation
NOV	Notice of Violation
NRC	Nuclear Regulatory Commission
RO	Reactor Operator
RP	Radiation Protection
RPA	Radiologically Protected Area
SFP	Spent Fuel Pool
SS	Shift Supervisor
SV	Solenoid Valve
TS	Technical Specification
UE	Unusual Event
URI	Unresolved Item
VIO	Violation