UNITED STATES NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION III

Report of Operations Inspection

IE Inspection Report No. 050-155/75-05

Licensee: Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49210

Big Rock Point Nuclear Plant
Charlevoix, Michigan

License No. DPR-6
Category: C

Type of Licensee: BWR (GE), 240 MWT

Type of Inspection: Program, Announced

Dates of Inspection: March 5-7 and 24-26, 1975

Dates of Previous Inspection: February 17-20, 1975 (Construction)

Principal Inspector: E L Jordan for D. W. Kiley 5/16/75 (Date)

Accompanying Inspector: L. R. Hunter 5/16/75 (Date)

Other Accompanying Personnel: C. M. Erb

L. R. Hueter

E. L. Jordan

Reviewed By: E. L. Jordan
Senior Reactor Inspector
Operations Branch
5/16/75 (Date)

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SUMMARY OF FINDINGS

Enforcement Action

The following items of noncompliance were identified during the inspection.

Violations: None.

Infractions

Contrary to 10 CFR 50.59, written safety evaluations were not performed as required or were inadequate for certain changes to the facility. Examples of these activities include:

1. Replacement of containment ventilation isolation valve CV-4097 with a different valve and actuator. (Paragraph 1.p, Report Details)

2. A change to operate with the inner door of the containment building emergency escape lock open. (Paragraph 6, Report Details)

3. A change in the fire protection system operating pressure and relief valve setpoint. (Paragraph 2.b, Report Details)

This infraction had the potential for causing or contributing to an occurrence related to health and safety.

Licensee Action on Previously Identified Enforcement Matters

None inspected.

Other Significant Findings

A. Current Findings

The reactor has remained shutdown since January 16, 1975 for correction of deficiencies in the design quality for instrumentation for the post incident cooling system.

B. Unresolved Items

1. Containment testing as required by Technical Specification 3.7 and as performed by CP does not appear to meet all the requirements of Appendix J to 10 CFR Part 50. Specifically:

   a. The 24 inch diameter gasket sealed flange on the containment side of CV-4097 which was last tested during the ILRT at 13 psig in April, 1974 after replacement on April 10, 1974 has not been subjected to a test at Pa (23 psig) as required by Appendix J. (Paragraph 1.p(5), Report Details)

   b. The 24 inch diameter butterfly type seat of CV-4097 has not been tested with pressure applied in the same direction as that of DBA conditions as required by Appendix J. (Paragraph 1.p(5), Report Details)

   c. The containment building emergency escape lock is leak tested at 5 psig instead of Pa (23 psig) as required by Appendix J due to design of the inner door which will not permit a greater reverse pressurization. (Paragraph 6, Report Details)

2. The containment sphere ventilation system does not appear to be designed to provide the vacuum relief function in conjunction with certain single failures of components of that system. (Paragraph 2.e (3), Report Details)

3. Replacement of containment isolation system switches (dps/9051 and 9052) with the original Mercoid switches may not provide the desired level of reliability. (Paragraph 1.n, Report Details)

C. Status of Previously Reported Unresolved Items: No change.

Management Interview

A management meeting was held with Messrs. C. J. Hartman, C. R. Abel, and D. E. DeMoor and other plant staff on March 7, 1975 to review findings concerning maintenance activities. Other findings were reviewed with Mr. C. R. Abel on March 26, 1975, and the above Summary of Findings were reviewed by Mr. Kiley with Mr. Hartman on May 5, 1975.

A. <u>Maintenance Activities (Reviewed March 7, 1975)</u>

1. The inspector stated that the failure to identify certain safety related activities as facility changes which were performed as maintenance items was in noncompliance with 10 CFR 50.59. (Summary of Findings - Infraction A) Licensee representatives acknowledged the statement.

2. The inspector stated that a number of procedures contained sections which were marked as "Not Applicable" but did not have the indicated review and signatures as required by the Administrative Procedures and is considered a deviation from established procedures and practices. The licensee acknowledged the above statement. (Paragraph 1.q.1, Report Details)

3. The inspector stated that in several procedures, check marks were utilized instead of initials which allow tracking for Quality Control purposes. The licensee acknowledged the above statement, and stated that the more recent procedures require initials and signatures. (Paragraph 1.q.2, Report Details)

4. The inspector stated that it appeared that many of the safety related maintenance procedures required quality control "Hold Points" but were not being indicated. The licensee acknowledged the above statement and indicated the problem would be addressed in the new Quality Assurance Program. (Paragraph 1.q.3, Report Details)

5. The inspector stated that the classification of systems relative to safety related maintenance appeared to need added emphasis. The licensee acknowledged the above statement.

6. The inspector stated that specific guidelines were needed in order to establish the level of activity which requires approved maintenance procedures. The licensee acknowledged this statement and indicated that the subject will be addressed in the new Quality Assurance program.

7. The inspector stated that he had reviewed the Preventative Maintenance programs with maintenance, and the Equipment Rotation program with operations. In both cases a mechanism for review to close out the Preventative Maintenance items does not exist. The licensee's representative acknowledged the above statement. (Paragraphs 1.q.6 and 1.q.7, Report Details)

8. The inspector stated that the maintenance activity concerning the Containment Isolation System pressure switches was not performed in accordance with facility change procedures because the activity was not considered a modification. (Paragraph 1.n, Report Details) The inspector stated that this item was considered unresolved pending a determination of suitability of the reinstallation of the Mercoid switches.

B. Design Activities (Reviewed March 26, 1975)

1. The inspector stated that the failure to perform written safety evaluations for several completed facility change packages was considered to be in noncompliance with 10 CFR 50.59. (Summary of Findings - Infraction A) The licensee acknowledged the above statements.

2. The inspector stated that failure to present the entire facility change packages (including all design requirements, all procedures and procedure changes, and detailed drawings) to the Plant Review Committee for review before the modification commences is considered a deviation from ANSI N45.2. The licensee acknowledged the above statement. (Paragraph 2.h.1, Report Details)

3. The inspector stated that failure to perform a timely review of completed facility change packages is a deviation from the established procedures and the intent of the design control program. The licensee acknowledged the above statement. (Paragraph 2.h.2, Report Details)

4. The inspector discussed with the licensee the specific activities following the replacement of the Containment Isolation System valve (CV-4097) and pointed out specific and general problems which occurred subsequent to the modification. The licensee acknowledged the above discussion. (Paragraph 1.p, Report Details)

C. Other Findings Reviewed (Reviewed March 26, 1975)

1. The inspector discussed the lack of engineering specifications at the facility and requested that the licensee determine the availability of the various plant specifications and consider making such specifications available at the facility for the purpose of inspection. (Paragraph 5, Report Details)

2. The inspector reviewed findings concerning containment penetration leak tests for the ventilation system and the escape lock which do not appear to meet the regulatory requirements of 10 CFR 50, Appendix J. (Paragraphs 1.p(5) and 6, Report Details)

REPORT DETAILS

Persons Contacted

Consumers Power Company

C. R. Abel, Operations Engineer
D. E. DeMoor, Technical Engineer
J. J. Fremeau, Associate Engineer
C. J. Hartman, Plant Superintendent
J. L. Krumin, Associate Engineer
D. G. LaCroix, Auxiliary Operator
S. E. Martin, General Engineer
E. McNamara, Shift Supervisor
R. A. McVay, Maintenance Supervisor
E. F. Peltier, Assistant Shift Supervisor
R. E. Schrader, Instrument and Control Supervisor
R. W. Voll, Reactor Engineer
J. J. Zabritski, Quality Assurance Engineer

1. Maintenance Activities

   The following safety related maintenance activities were selected for review:

   a. Steam Drum (RSD) Dated April 8, 1974

      (1) The repair and repacking of the west drum level sensor root valve was accomplished.

      (2) No discrepancies were noted.

   b. Reactor Vessel (RVG) Dated April 21, 1974

      (1) The installation of the reactor vessel head was completed.

      (2) No discrepancies were noted.

   c. Post Incident System (PIS) Dated May 1, 1974

      (1) The pressure switch, 1G11F in the core spray system required repair due to internal leakage.

      (2) The required procedures, materials, and instructions utilized were attached.

      (3) No discrepancies were noted.

d. Emergency Power System (EPS) Dated May 17, 1974

(1) The emergency diesel battery charger required a relay (contact) repair which had caused an overcharging condition.

(2) No discrepancies were noted.

e. Reactor Coolant System (RCS) Dated June 15, 1974

(1) The seal on the No. 2 recirculation pump was removed and replaced.

(2) That attached maintenance procedure was not completed. Step 4.5.7, the completed procedure review, and the QA review were not signed.

(3) No other discrepancies were noted.

f. Main Steam System (MSS) Dated July 11, 1974

(1) The limitorque operator on valve MO-N004 was inspected and lubricated.

(2) No discrepancies were noted.

g. Main Steam System (MSS) Dated July 13, 1974

(1) The limitorque operator on valve MO-7050 was inspected and lubricated.

(2) No discrepancies were noted.

h. Emergency Cooling System (ECS) Dated July 15, 1974

(1) The limitorque operators on valves MO-7062 and MO-7063 were inspected and lubricated.

(2) No discrepancies were noted.

i. Post Incident System (PIS) Dated July 15, 1974

(1) The core spray valve MO-7051 was dismantled and inspected.

(2) No discrepancies were noted.

j. <u>Reactor Cleanup System (RCS) Dated July 18, 1974</u>

(1) The rear bearing on the cleanup pump was replaced.

(2) No discrepancies were noted.

k. <u>Control Rod Drive System (CRD) Dated February 17, 1975</u>

(1) Time delay relay 4K3 was replaced after failure.

(2) No discrepancies were noted.

l. <u>Control Rod Drive System (CRD) Dated December 16, 1974</u>

(1) Control rod drive assembly No. 252 was dismantled, inspected, cleaned, repaired and reassembled.

(2) No discrepancies were noted.

m. <u>Emergency Power System (EPS) Dated November 22, 1974</u>

(1) The fuel oil supply pump to the emergency diesel was replaced with a later model pump.

(2) The activity was completed with no apparent discrepancies.

n. <u>Containment Isolation System (CIS) Dated February 6, 1975</u>

(1) The previously installed (April 4, 1973) snap action switches (dps/9051 and dps/9052) were replaced with the original Mercoid switches.

(2) The maintenance activity was completed with no apparent discrepancies.

(3) The replacement of the snap action switches (Barksdale) with the originally installed (Mercoid) swithces was accomplished as a maintenance activity and not included in the facility change program. The originally installed Mercoid switches were previously removed in early 1973,[1] and replaced at that time with the Barksdale switches.

The basis for not doing facility change and written safety review, according to the Licensee's representatives, was the fact that the switches were original equipment and therefore required no written safety review.

1/ Ltr CP to DL, dtd 5/8/73 (AD-5-73).

o. Emergency Power System (EPS) Dated March 14, 1974

(1) The emergency generator voltage output indication failure on the local auxiliary meter panel was corrected by replacing a potential transformer with a control transformer and the voltmeter circuit was modified to provide proper voltage indication.

(2) The associated maintenance activity was completed with no apparent discrepancies.

(3) The activity is identified as marginal as a maintenance activity based upon the licensee's Quality Assurance Procedures and Administrative Procedure 7.2.

p. Containment Isolation System (CIS) Dated April 10, 1974

CV-4097 (Containment Ventilation Supply Isolation Valve) and the pneumatic operator were replaced with components having different design and operating characteristics than the original installation. This equipment modification was accomplished using a maintenance procedure. However, this change had not been designated a facility change per the Procedures Manual, Section A, 7.2.3 and an engineering evaluation of the modification and post maintenance testing requirements pursuant to 10 CFR 50.59 apparently had not been adequately completed prior to installation of the new valve. The following was found relative to the valve and actuator modification.

(1) Installation and testing requirements were physically completed on April 10, 1974, per maintenance procedure MCIS-1, Revision 0, dated March 26, 1974.

(2) Testing per the above maintenance procedure found that closure times for CV-4096 and CV-4097 were 9 seconds and in excess of the Technical Specification (TS 3.4.3(f)) maximum closure time of 6 seconds. This was due to the additional volume of the CV-4097 valve operator modificati The closure times of the valves were made less than 6 seconds by isolating one the three volume cylinders associated with the new valve operator. (Paragraph 2.e, Report Details)

(3) The modification to remove one of three volume cylinders from the valve operator was subsequently designated a facility change (C-254) and engineering review found that

the manufacturer did not recommend this modification. The volume cylinder was restored per a facility change (C-258) and a temporary Technical Specification change for a maximum closure time of 10 seconds was obtained 2/. During the June - July 1974 refueling the longer closure times were corrected to within 6 seconds by installing larger capacity pneumatic control valves. (Paragraphs 2.e and f, Report Details)

(4) The licensee found that the increased volume of the actuator assembly resulted in the backup pneumatic supply, consisting of one nitrogen bottle, not having sufficient capacity to provide 50 valve operating cycles as required by the FHSR, Section 3.5.2.3 for the post DBA vacuum breaker function of the ventilation supply valves. This was corrected by adding three additional nitrogen bottles (facility change C-260) and by functionally testing the backup supply capacity during the June - July 1974 refueling outage.

(5) During the ILRT on April 24, 1974 the licensee found that CV-4097 containment side flange leaked excessively. The leakage was corrected by tightening the flange bolts and the flange was tested as part of the ILRT boundary at a pressure of 13 psig. During the ILRT it was found that valve CV-4097 (butterfly type) seat leaked so that CV-4097 (check type valve) held the test pressure. 3/ 4/ 5/ 6/

The gasket-sealed flange has not been tested at the design basis accident pressure (Pa) in accordance with 10 CFR 50, Appendix J, Sections III B.2 and IV.A. (Pa is 23 psig per the FHSR, Section 3.2.1). In addition the butterfly type valve seat has not been tested with a pressure of Pa applied in the direction of accident conditions per Appendix J, Sections III.C.1 and III.C.2.

q. <u>Other Findings</u>

Other items discussed with the Licensee's representative are:

(1) Certain maintenance procedure sections, including recirculation pump seal activity, dated June 15, 1974, were not completed. 7/ This is considered a deviation with Administrative Procedure 8.0(B.3.1) which specifies procedure adherence and temporary change methods.

2/ Ltr CP to DL dtd 5/2/74.
3/ AO-10-74 Rpt, 5/6/74 (CV-4097 flange leakage).
4/ Ltr CP to DL, 8/2/74 (Rpt of ILRT).
5/ RO Inspection Rpt No. 050-155/75-04.
6/ Ltr CP to DL, 9/23/74 (Correction to Rpt ILRT).
7/ Ltr CP to DL, 7/23/74, AO-22-74.

(2) In certain procedures check marks, instead of initials, were used to indicate completed steps. The subject procedures indicated that initials were to be used to document completion of certain procedural steps.

Administrative Procedures 7.0 and 8.0 require detailed written procedures with the appropriate signoffs.

(3) Quality control "Hold Points" have not been applied to safety related maintenance procedures.

Administrative Procedures 7.0 and 8.0 require detailed procedures and checklists and ANSI N18.7 Section 5.1.6 requires detailed procedures commensurate with the activity with means of assuring quality.

2. Design Activities

The following safety related facility change activities were selected for review.

a. Waste Gas System (WGS) Dated July 1, 1974

(1) The off-gas flow transmitter was replaced with a new type (Bailey with a Foxboro) because of low flow inaccuracies.

(2) The associated maintenance activity was completed with no apparent discrepancies.

(3) Although the review performed addressed the original equipment model performance, the original design specifications with documented acceptance criteria and test requirements were not addressed.

b. Fire Protection System (FPS) Dated February 19, 1975

(1) The fire system pressure switch (PS-609-2) was reset from 95 psig increasing, to 90 psig increasing and the accumulator relief valve (RV-5040) setpoint was reduced from 110 psig to 95 psig.

(2) The associated maintenance activity was completed with no apparent discrepancies.

(3) The fire protection system appears to be safety related in that it is a source of water for ECCS functions and it is designated as a QA Category System per the Operational Quality Assurance Procedures Manual, Section 3. This facility

change was not designated safety related in accordance with the Procedures Manual, Section A, 7.2.3 and there was not a documented safety ev luation of the change pursuant to 10 CFR 50.59 regarding the effect of the reduced pressure for ECCS functions.

c. Fuel Handling System (FHS) Dated February 14, 1974

(1) The sipper can vacuum cleaning system was fabricated to clean the apparatus following dry-sipping of an irradiated fuel assembly for failed fuel identi cation.

(2) The associated maintenance activity was completed with no apparent discrepancies.

(3) Weaknesses were identified in the details provided regarding the consequences of the operations; including the lack of design specifications, codes, acceptance tests, and acceptance criteria in order to certify the system prior to placing in operation.

d. Control Rod Drive System (CRD) Dated April 5, 1974

(1) The bottom rollers were removed from all peripheral control rods due to vibration and subsequent pin failures.

(2) No apparent discrepancies were noted.

e. Containment Isolation System (CIS) Dated April 17, 1974

(1) One of three air cylinders on the containment vent isolation valve (CV-4097), was removed in the field to meet the closing time criteria as stated in the associated maintenance procedure.

(2) The activity was not performed in accordance with facility procedures since the activity was performed on April 10, 1974, and the facility change (C-254) was not initiated until April 17, 1974.

(3) Based upon review of the vacuum breaker function of the containment isolation system, the inspector found that possible single failure modes exist. The apparent failure modes include: failure of either ventilation supply valve to open, and failure of the single vacuum switch (dps/9051) to actuate.

f. Containment Isolation System (CIS) Dated July 8, 1974

(1) The solenoid valves (SV-9151 and SV-9152) on the containment ventilation supply isolation valves (CV-4096 and CV-4097) were replaced to provide a large path for operating air flow to the valves (facility change C-259).

(2) No apparent discrepancies were noted.

g. Primary Coolant System (PCS) Dated April 6, 1974

(1) The gearing in the limitorque operators for the recirculation pumps discharge valves was replaced to decrease the opening times to be within General Electric recommended values, Technical Specifications, and Safety Analysis Report requirements[8].

(2) No apparent discrepancies were noted.

h. Other Findings

Additional items discussed with Licensee's representatives are:

(1) At the time of review certain facility change packages did not have all of the associated information attached in order to allow a meaningful PRC review to assure all requirements of 10 CFR 50.59 and the Quality Assurance Manual were being met. This associated information includes special procedures, surveillance testing, and operating procedures changes, and detailed sketches of the change. This item is considered a deviation from the ANSI N45.2, Section 4.

(2) In a number of facility change packages the subsequent review of the completed packages, as required by the administrative procedures and quality assurance procedures, was not performed in a timely manner. In certain instances this review by the operations engineer, quality assurance engineer, and others did not occur until 4 to 6 months after the completion of the work. This period of time in most cases extended into a plant operating phase which required the equipment or components to be operable. This practice is considered to be in deviation with the facility change procedure 7.2 and ANSI N45.2 (Section 7) as established to comply with document control measures.

(3) Activities involving the fire system relief valve setpoint change, the off-gas flow transmitter replacement, and certain other facility change packages, the total package was performed by one person. This includes the modification,

8/ Ltr CP to DL, UE-R-6-73.

the review and the implementation of the facility change. This appears to be in deviation with the design control program as indicated by ANSI N45.2, (Section 4) requiring independent design review of safety related modification and changes.

4. Containment Supply Vent Leakage (AO-8-75)

During the March 1975 ventilation supply penetration testing it was found that CV-4097 valve seat leaked excessively with pressure applied in the reverse direction from DBA conditions (AO-8-75).[9/]

The inspector found that the licensee was in the process of adjusting and retesting valve CV-4097 on March 27, 1975. The licensee's representative stated that information obtained from the valve vendor indicated that the rubber seat of the valve is apparently intended for steam applications with a wet environment and that the shelf life of the rubber in a dry atmosphere is one year.

5. Plant Records

Inspection review found that neither detailed system design specifications nor detailed engineering specifications were available at the facility [illegible] two exceptions: the Final Hazards Summary Report (a summary [illegible]) and a piping specification summary list prepared by the archi[illegible] engineer. It appears that such detailed specifications should be used as a basis for facility engineering review of proposed design changes pursuant to 10 CFR 50.59, facility changes and modifications, and quality assurance for procurement of components and material. The licensee representatives stated that it was not known if such records existed nor where they are located. The licensee indicated that the availablity of these records would be investigated.

In addition, records from the construction of the facility such as inspection records and "as-built" records do not appear to be available at the facility. For example it was not known whether original inspection records or radiographs existed for inspection review of AO-5-75 (Steam Line Weld Defect) and AO-7-75 (Emergency Condenser Outlet Pipe Defect).[10/] The licensee representative indicated that some NDT records from plant construction are known to have existed however their availability and location was not known. It was also noted by the inspector that isometric drawings of piping installations were not available.

9/ AO-8-75 Rpt, 3/27/75.
10/ IE Inspection Report No. 050-155/75-02.

6. Emergency Escape Lock Leak Testing

A change to the normal operating condition of the containment emergency escape lock had been effected in May 1974.[11] Prior to this change the inner door of the lock had been maintained closed by operating practice. The inner door had been the pressure boundary for the last containment integrated leak rate test (ILRT) of April 24-29, 1974. Inspection review found the following:

a. The change to open the escape lock inner door had been made for personnel safety reasons to facilitate escape in the event of a safety valve actuation and discharge of steam to the containment. The inspector found that the inner door of the escape lock had a lever operated mechanism and that the door could be opened from a fully shut condition in less than 5 seconds with little effort.

b. The change to open the escape lock inner door had been reviewed by the Plant Review Committee (PRC) with respect to personnel safety based upon the probabilities for actuation of the safety valves and for actuation the ADS system to be installed. However, according to the licensee and available records the change had not been reviewed pursuant to 10 CFR 50.59 based upon the leakage testing requirements of Appendix J to 10 CFR 50 and the reliability of the containment integrity. The change had been implemented in the Procedures Manual, Section A, 6.1 by a change dated May 1, 1974.

c. The escape lock outer door and associated equalizing valve had been leak tested on March 19, 1974 at 5 psig and on September 19, 1974 at 4.6 psig by pressurizing the lock.[12] The inner door and associated equalizing valve had been leak tested at 13 psig as a containment boundary during the ILRT of April 1974,[13] however, the outer boundries of the lock had not been pressurized.

The change to open the inner door resulted in containment boundaries (the outer door and associated equalizing valve) which had not been tested as part of the ILRT per Appendix J, Section III, A.1(d). In addition the escape lock had not been tested to the design basis accident (DBA) pressure (Pa) of 23 psig per Appendix J, Section III, B.2 and Section III, C.2.

The licensee representative stated that the escape lock had only been tested to 5 psig because past experience had found that the lock could not be tested to a higher pressure because of difficulty in restraining the inward opening of the inner door and the resulting leakage from the lock into the containment.

11/ RO Inspection Rpt No. 050-155/74-10.
12/ Test procedure, T 180-1, Section 4.4, Escape Lock Leak Rate Test (semi-annual test).
13/ Test procedure T-730-01, Containment Integrated leak Rate Test, dtd 4/19/74.

7. Facility Inspection Tours

The inspectors toured the control room area, auxiliary equipment area, turbine building and the containment building, including the recirculation pump room and the CRD room. No significant discrepancies or housekeeping items were found. The cable penetration areas inside and outside containment were inspected on March 27, 1975 and no fire hazards were found.