**Consumers Energy**

A CMS Energy Company

Big Rock Point Nuclear Plant
10269 US-31 North
Charlevoix, MI 49720

**Kenneth P. Powers**
Plant General Manager

May 9, 1997

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

**DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT PLANT - LICENSEE EVENT REPORT 97-004: POTENTIAL LOSS OF DC POWER FOR THE PRIMARY CONTAINMENT SPRAY AND LIQUID POISON SYSTEMS.**

Licensee Event Report 97-004, **POTENTIAL LOSS OF DC POWER FOR THE PRIMARY CONTAINMENT SPRAY AND LIQUID POISON SYSTEMS,** is attached. This event is reportable to the Nuclear Regulatory Commission in accordance with 10 CFR 50.73(a)(2)(v) - "Any event or condition that alone could have prevented the fulfillment of a safety function of structures or systems".

Robert J Addy
Plant Manager

1/1
IE22

CC: Administrator, Region III, USNRC
NRC Resident Inspector - Big Rock Point

ATTACHMENT

160045

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503

# LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4)
Potential Loss of DC Power for the Primary Containment Spray and Liquid Poison Systems.

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| 04 | 10 | 97 | 97 | 004 | 000 | 05 | 09 | 97 | FACILITY NAME | DOCKET NUMBER |

OPERATING MODE (9): N

POWER LEVEL (10): 0%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| | 20.2201(b) | | 20.2203(a)(2)(v) | | 50.73(a)(2)(i) | | 50.73(a)(2)(viii) |
| | 20.2203(a)(1) | | 20.2203(a)(3)(i) | | 50.73(a)(2)(ii) | | 50.73(a)(2)(x) |
| | 20.2203(a)(2)(i) | | 20.2203(a)(3)(ii) | | 50.73(a)(2)(iii) | | 73.71 |
| | 20.2203(a)(2)(ii) | | 20.2203(a)(4) | | 50.73(a)(2)(iv) | | OTHER |
| | 20.2203(a)(2)(iii) | | 50.36(c)(1) | X | 50.73(a)(2)(v) | | Specify in Abstract below or in NRC Form 366A |
| | 20.2203(a)(2)(iv) | | 50.36(c)(2) | | 50.73(a)(2)(vii) | | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (INCLUDE AREA CODE) |
|---|---|
| Michael D. Bourassa, Licensing Supervisor | 1-616-547-8244 |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS |
|---|---|---|---|---|---|---|---|---|---|
| B | EI | BKR | G080 | N | | | | | |
| | | | | | | | | | |

| SUPPLEMENTAL REPORT EXPECTED (14) | | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|---|
| | YES (If yes, complete EXPECTED SUBMISSION DATE). | NO | | | | |


ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On March 19, 1997, a load test was conducted on feeder breaker 72-12, DC Distribution Panel #1 as part of an action plan to verify breaker performance. The plant was in the cold shutdown condition. Following what was believed to be a successful performance of the magnetic load test, breaker 72-12 would not manually reset. Breaker 72-12 was opened, and verified to contain only a magnetic trip feature. Plant drawings indicated that it should have a thermal trip feature. On April 10th, 1997, @ approximately 1600, the Big Rock Point Staff concluded from an additional calculation that a loss of selective coordination over a minor portion of the interface between breaker 72-12 and other DC breakers in the circuit could have resulted in a loss of the panel due to a nonsafety-related fault. The safety-related loads supplied by the panel that could have been affected are MO-7064, Primary Containment Spray valve, and the Liquid Poison System squib valve control circuits.

The root cause for this event is an inadequate design control program prior to 1976. No design review program covered the physical verification of this breaker

Breaker 72-12 was removed and replaced with a load-tested, thermal-trip-only spare breaker. Long term corrective actions include the performance and development of preventative maintenance tasks to remove and test the DC breakers associated with the panel.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

IDENTIFICATION OF EVENT

This event is reportable to the Nuclear Regulatory Commission pursuant to:

10 CFR 50.72(b)(2)(iii) - Licensees shall report: "Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(A) Shutdown the reactor and maintain it in a safe shutdown condition;
(C) Control the release of radioactive material; or
(D) Mitigate the consequences of an accident.

10 CFR 50.73(a)(2)(v) - Licensees shall report: "Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(A) Shutdown the reactor and maintain it in a safe shutdown condition;
(C) Control the release of radioactive material; or
(D) Mitigate the consequences of an accident.

REFERENCES

C-BRP-97-0194; Loss of Selective Coordination and Configuration Control for 72-12, DC Distribution Panel (DO2) Feeder.

EVENT DESCRIPTION

The Big Rock Point Staff reviewed INPO Significant Event Report (SEN) 149, Insufficient Testing of Molded Case Circuit Breakers [BKR] Affects Safety System Operability issued January 29, 1997. This SEN was related to the problems involving DC molded circuit case breakers discovered at Consumers Energy Company's Palisades Nuclear Power Plant. This review and evaluation lead to the conclusion that Big Rock Point Plant's DC breakers could be similarly susceptible. On February 28, 1997, plant management reviewed the situation with plant engineering, and accepted their recommendation that continued plant operation would be acceptable based on reasonable assurance that the breakers could provide their safety function.

On March 2, 1997, a forced outage due to a steam leak on a core spray system valve removed the plant from service. This outage afforded the opportunity to functionally test a sample of the DC breakers of concern. Several breakers in a DC panel and their feeder breakers were tested. All breakers were acceptable, except for breaker 72-12, DC Distribution Panel [PL] #1 (DO2)

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On March 19, 1997, Consumers Energy's Laboratory Services conducted a load test on feeder breaker 72-12. Following what was believed to be the successful performance of the magnetic load test, breaker 72-12 would not manually reset. The Big Rock Point staff questioned why a magnetic load test was conducted when the plant instrument drawings indicated that the breaker only required a thermal trip. Laboratory Services referred the Big Rock Point staff to the setting sheet, which identified breaker 72-12 as having both magnetic and thermal protective trip features. When both trip features are present, Laboratory Services usually performs the magnetic trip test first because of the higher rate of failure of the thermal test. At this point, breaker 72-12 was opened, and verified to contain a magnetic trip feature only. The impact of having a magnetic only vs. thermal only feature was evaluated by engineering. Initially, the minor area of coordination concern was acceptable due to cable sizing and length. However, on April 10th, 1997, @ approximately 1600, the Big Rock Point Staff concluded from a calculation provided by Consumers Energy System Protection Engineering, that a minor loss of selective coordination of breaker 72-12 and other breakers in the circuit could have resulted. This condition could exist if the emergency back-up lights [LF] for the stack were in service supplied by breaker 72-1D44 through the DC Distribution Panel (D02) and a fault occurred. Panel 7L is located within the fault distance that can affect the DC Distribution Panel (D02). The fault distance was determined by Consumers Energy System Protection Engineering to be less than 383 feet for (#1/0) cable. During a Loss of Cooling Accident (LOCA) this non-EEQ DC circuit *could have* caused breaker 72-12 to open on mis-coordination de-energizing the 125 VDC System [EI]. The safety-related loads supplied by the panel that could have been affected are MO-7064, Primary Containment Spray valve, and the Liquid Poison System [BR] squib valve control circuits.

ANALYSIS

During April 1961, General Electric, Bechtel Corporation, and Consumers Power documented that 125 Volt DC breaker 72-12 was a *thermal only trip* type of application. Sometime between 1962 and 1976, breaker 72-12 was changed to a *magnetic only trip*. A review of the Big Rock Point Maintenance Journal and the Advanced Maintenance Management System (AMMS) revealed that breaker 72-12 has been in place since 1976. If there were any maintenance activities associated with breaker 72-12 during the 1962 to 1976 window of time, they were not documented. Interviews with plant personnel indicate that breaker 72-12 probably did not receive any associated preventative maintenance during this time. However, the breaker could have been changed out and not documented under any controlled process while performing corrective maintenance. Consumers Energy System Protection Engineering was contacted to validate this fact. System Protection Engineering could not dispute the proposed cause because their group had not come into existence until the 1970s. It should be noted that many Consumers Energy personnel believe that the feeder breaker has been in place since original construction.

The extent of this condition affecting any other major DC breakers has been evaluated. The only other potentially affected DC breaker is 72-11, 125 VDC Motor Control Center Feeder. The incoming line breaker 72-11 was load tested on April 2, 1997.

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INDUSTRY EXPERIENCE

Industry experience was reviewed for cases dealing with breaker miscoordination. Two NRC Information Notices 88-45 and 92-29 discuss this topic. Industry experience was also reviewed for cases associated with breaker testing. NRC Information Notices 93-64, Periodic Testing and Preventive Maintenance of Molded Case Circuit Breakers, was issued August 12, 1993. At that time, the station's review was not deep enough to verify and validate that the 125 VDC safety-related breakers were captured in the PM program and maintenance was being performed. The PM validation did identify these breakers as outliers and Periodic and Predetermined Activity Control (PPAC) development was recommended. The issue was revisited by SEN 149, Insufficient Testing of Molded Case Circuit Breakers Affects Safety System Operability, issued by the Institute of Nuclear Power Operations (INPO) on January 29, 1997. If Big Rock Point had investigated the issues more thoroughly, this condition would have been discovered earlier.

MAINTENANCE RULE

Breaker 72-12 is within scope of the Maintenance Rule. The Emergency Power System Maintenance Rule function (EPS-04) is to provide electrical power to safety related equipment. The Station Battery is within the Emergency Power System and breaker 72-12 is a Station Power System interfacing component.

ROOT CAUSE

The root cause of this event was determined to be inadequate design control during original plant construction and/or early plant operating life (~ 1962 to 1976).

Furthermore, the failure to validate the design function of the 125 VDC system was caused by management insensitivity to plant and industry experience relative to the rising standards of design basis verification and testing since 1962.

CORRECTIVE ACTION

Immediate

1. Breaker 72-12 was removed, and replaced with a correctly-sized, load-tested, thermal-trip-only tested spare breaker.

2. The incoming line breaker, 72-11, was successfully load tested on April 2, 1997.

3. The current design control processes have been revised and improved many times to capture the present industry standards since 1976.

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Long Term

1. Breakers 72-11 through 72-44 which are associated with the DC Distribution Panel (D02) Feeder will be tested.

   **THIS ACTION WILL BE COMPLETED BY THE END OF THE 1997 REFUELING OUTAGE.**

2. Evaluate and document that all safety-related AC and DC breakers installed in the plant are within the plant design.

   **THIS ACTION WILL BE COMPLETED BY END OF THE 1997 REFUELING OUTAGE.**

3. Review open Preventative Maintenance (PM) Validation outliers and determine if other PM recommendations have not been implemented.

   **THIS ACTION WILL BE COMPLETED BY JUNE 15, 1997.**

SAFETY SIGNIFICANCE

Loss of DC (Station Battery) is an analyzed Limiting Condition of Operation (LCO), and in accordance with technical specifications requires the initiation of a plant shutdown. Other analyzed accidents and the mitigating requirements of systems addressed in the Technical Specifications can be supported assuming a loss of the station battery (i.e. Core Spray can be provided via AC valves and either diesel or motor driven fire pump). Appendix R Safe Shutdown Analysis required that a breaker coordination study be performed to provide reasonable assurance that plant shutdown could be achieved following a fire in all plant areas. The study identified similar coordination issues, the more significant coordination problems were corrected by modification, and the minor issues were determined to be acceptable as is.

The incorrect breaker type was installed for breaker 72-12 which resulted in a fault protection coordination problem with a single breaker in panel D02. The coordination problem involved only a small overlap of the breaker curves so its affect would have been limited. Therefore, this event was of limited safety significance for the facility.