Consumers Power
POWERING MICHIGAN'S PROGRESS
Big Rock Point Nuclear Plant, 10269 US-31 North, Charlevoix, MI 49720

Patrick M Donnelly
Plant Manager

April 4, 1994

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT PLANT - LICENSEE EVENT REPORT 94-004; UNPLANNED ACTUATION OF AN ENGINEERED SAFETY FEATURE - DIESEL FIRE PUMP.

LICENSEE EVENT REPORT 94-004; **UNPLANNED ACTUATION OF AN ENGINEERED SAFETY FEATURE - DIESEL FIRE PUMP,** is attached. This event is reportable to the Nuclear Regulatory Commission pursuant to 10 CFR 50.72(b)(2)(ii) and 10 CFR 50.73(a)(2)(iv).

Patrick M Donnelly

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

ATTACHMENT

IE22 1/1

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
BIG ROCK POINT PLANT	0 5 0 0 0 1 5 5	1 OF 0 3

TITLE (4)
UNPLANNED ACTUATION OF AN ENGINEERED SAFETY FEATURE - DIESEL FIRE PUMP

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 NAMES	DOCKET NUMBER(S)
0 3	0 8	9 4	9 4	0 0 4	0 0	0 4	0 4	9 4	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

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

20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Michael D Bourassa, Senior Licensing Engineer	616 547-6537

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	KP	CNTR	A160	N					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) — X NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

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

On March 8, 1994, the reactor was in the cold shutdown condition and steam drum instrument lines were being backfilled to equalize level instrument readings. Both the Electric and Diesel Fire Pumps (these fire pumps also function as low pressure core spray pumps) were inhibited from starting (per procedure) by handswitches in the control room. At approximately 1253, the Diesel Fire Pump started unexpectedly.

The root cause identified a "loss of voltage indication" in the diesel fire pump handswitch contacts. The voltage feeds the fire pump logic module to disable the auto start feature. The loss of voltage also contributed to the malfunction of an associated optical isolator/light; however additional troubleshooting identified a problem with socket-to-pin fit.

The handswitch contacts were changed, and the optical isolator/light socket fit was corrected. The handswitch was declared operable at 2306 hours that same day. Additional corrective action included an investigation of other safety-related optical isolator/light socket-to-pin fits before exiting cold shutdown.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
BIG ROCK POINT PLANT	0 5 0 0 0 1 5 5	9 4	— 0 0 4 —	0 0	0 2 OF 0 3

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

IDENTIFICATION OF EVENT

I. Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System. However, actuation of an ESF, including the RPS, that resulted from and was part of the preplanned sequence during testing or reactor operation need not be reported. This event is reportable because the actuation of the diesel fire pump was not part of a pre-planned sequence.

References

a. 10 CFR 50.72(b)(2)(ii), and

b. 10 CFR 50.73(a)(2)(iv).

CONDITIONS PRIOR TO THE EVENT

The reactor was in the cold shutdown condition. The facility was shutdown March 2, 1994, for a scheduled maintenance outage.

DESCRIPTION OF THE EVENT

On March 8, 1994, the reactor [RX] was in the cold shutdown condition and the steam drum [SD] instrument lines were being backfilled with water to fine tune steam drum level instrument readings. This activity had the potential to generate an electric motor driven and diesel engine driven fire pump [MO-ENG; KP;P] engineered safety features start signal (low water level) through the Reactor Depressurization System (RDS) logic; therefore the pumps were inhibited by hand switches [HS] HS-7085 and HS-7086 on the C-40 panel [PL] in the control room [NA]. However, at approximately 1253 hours that same day, the diesel fire pump started unexpectedly.

NOTE: Both fire pumps function as low pressure core spray [BM] pumps. The source of water for the core spray system is supplied by the Fire Protection System [KP]. However, since the RDS is not required to be in an automatic mode when the reactor is in the cold shutdown condition, the fire pump hand switches were placed to "INHIBIT" in accordance with facility procedures to prevent the fire pumps from auto starting.

By 1431 hours, a four-hour report had been made to the NRC Operations Center, notifying the agency that an unplanned ESF actuation had occurred.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
BIG ROCK POINT PLANT	0 5 0 0 0 1 5 5	9 4	— 0 0 4	— 0 0	0 3 OF 0 3

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

CAUSE OF THE EVENT

The root cause of this event has been attributed to equipment failure. The contacts [CNTR] manufactured by Allen-Bradley Company [A160] for HS-7086 (Diesel Fire Pump AUTO/INHIBIT switch) were tested and failed to indicate voltage as designed. The voltage feeds the fire pump logic module [IMOD] L2 via a relay [91] in RDS Actuation Cabinet Number 3 to disable the auto start feature. The loss of voltage also contributed to the intermittent malfunction of an associated optical isolator/light [OB].

CORRECTIVE ACTION TO PREVENT RECURRENCE

Immediate Action

The contacts for HS-7086 were replaced and the handswitch was declared operable at 2306 hours that same day.

The problem with the intermittent operation of the optical isolator/light was determined to be a socket-to-pin fit. The pins were identified as being too close together, and not allowing sufficient contact for the socket. The pins were adjusted, and the optical isolator/light was tested satisfactory.

Prior to returning the facility to power operation, an investigation for similar optical isolators/lights that perform a related safety function was conducted, and no others were identified. A sampling of approximately 15% of the non-safety RDS optical isolators/lights was then conducted. No problems were encountered with socket-to-pin fit. No new problems were identified.

SAFETY SIGNIFICANCE

Engineered Safety Features are provided to mitigate the consequences of events, and therefore should work properly when called upon and should not be challenged unnecessarily. Even though equipment failure caused an unnecessary challenge, the pump worked properly by actuating when the RDS logic was satisfied by the loss of voltage in the circuit. The safety significance (invalid signal, reactor in cold shutdown, core spray valves remained closed and flow was never initiated) with regards to the conservative fire pump actuation is negligible.

OTHER REFERENCES

LER 91-005 dated July 15, 1991.
LER 92-007 dated May 26, 1992.
LER 93-007 dated August 12, 1993.
LER 94-003 dated March 28, 1994.