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Nuclear Power Plant
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Lycoming, New York 13093
315 342-3840

New York Power Authority

William Fernandez II
Resident Manager

November 19, 1990
JAFP-90-0834

United States Nuclear Regulatory Commission
Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 90-023-00
Manual Scram - Blocked Intake Screens

Dear Sir:

This Licensee Event Report is submitted in accordance with 10 CFR 50.73(a)(2)(iv).

Questions concerning this report may be addressed to Mr. Hamilton Fish at (315) 349-6013.

Very truly yours,

WILLIAM FERNANDEZ

WF:HCF:lar

cc: USNRC, Region 1
USNRC Resident Inspector
INPO Records Center
American Nuclear Insurers

Cert No P229764407

IE22

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)
JAMES A. FITZPATRICK NUCLEAR POWER PLANT	0 5 0 0 0 3 3 3	1 OF 0 5

TITLE (4) Manual Reactor Scram Due to Blocked Circulating Water Intake Screens Due to Loss of Differential Pressure Signal Due to Procedural Deficiency

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)
1 0	1 9	9 0	9 0	– 0 2 3 –	0 0	1 1	1 9	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 4 5

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
Hamilton C. Fish	AREA CODE 3 1 5 3 4 9 – 6 0 1 3

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
D	K E	S C N	J 0 3 3	Y					

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)

ABSTRACT

At 0912 on 10/19/90 the reactor was manually scrammed from 45 percent power as a conservative measure due to inability to clear the circulating water system traveling screens. The rate of supply of circulating water had been reduced due to clogging of the cooling water intake screens [KE] by leaves and lake weeds. One of three 50% capacity screens was out of service for scheduled preventive maintenance. A shift in wind direction contributed to an unusually large debris accumulation on the remaining two screens. The screen differential pressure alarm and screenwash systems, which would have provided early indication of fouling, had been unintentionally disabled during the maintenance of the out of service screen due to a procedural deficiency. Shear pins on the remaining two screens failed. It was necessary to remove two of the three main condenser circulating water pumps [KE] from service to clear and repair the screens. This necessitated the power reduction and subsequent manual scram. The plant returned to service at 0418 on 10/21/90 (43 hours, 6 minutes off line).

Appropriate steps have been added to procedures. The differential pressure instrumentation has been added to operator log sheets, plant flow diagrams, and instrument calibration schedules.

Description

All plant cooling water is obtained through a submerged intake structure from Lake Ontario. The water passes through three nominal 50% capacity large 3/8-inch mesh traveling screens which are designed to remove solid materials (lake weeds, leaves, fish, debris) from the water before it enters the intake bays for the circulating water pumps [KE], fire pumps [KP], and emergency [BI] and normal service water pumps [KG]. One of the three traveling screens [KE] was out of service for routine scheduled preventive maintenance. The screens may be operated in a manual continuous travel mode or in an automatic mode. In the automatic mode, screen rotation is initiated by either a four-inch differential level across the screens or by an interval timer for ten minutes once every four hours. Screenwash water is activated by these same signals and is required for screen operation.

The screens were operating continuously in the manual mode during most of the 1500 to 2300 shift on October 18, 1990. Because no abnormal accumulation of debris were observed in the screenwash collection basket, the system was switched to automatic prior to shift turnover. During the 2300 to 0700 shift on October 19th, the screenwash water was observed to be flowing with no unusual accumulation of debris in the screenwash collection basket at 0200 and 0600. Actual rotation of the screens was not verified.

The oncoming shift supervisor for the 0700 to 1500 shift observed a strong northerly (toward shore) wind condition as he entered the plant site. Based on these strong wind conditions he directed that the screens be placed in continuous operation and checked for incoming debris. At 0710 the two available traveling water screens were placed in manual operation. Although screenwash water was flowing, no rotation was observed at the chain drive. Repair efforts were started. The shear pins for both of the inservice screens were found to have failed.

At 0734 a main condenser high differential temperature alarm was received. Accordingly, an immediate reduction in plant power was commenced. The Superintendent of Power and the Assistant Operations Superintendent met at the screenhouse area and, observing the differential water level across the screens, advised the shift supervisor that removal of main circulating water pump C from service (one of three) would probably be required. The possibility of the need to manually scram the plant was discussed and communicated to the Operations Superintendent who had arrived in the control room.

As repair efforts continued, it became apparent that one of the three main condenser circulating water pumps would have to be removed from service. Accordingly, the reduction in power continued using a combination of control rod [AA] insertion and reduction of reactor water recirculation pump [AD] speed. During the power reduction, the observed difference (drop) in water level across the screens was observed to be about ten feet (ten inches of level difference was the alarm point for high differential level).

At 0820 the C main condenser circulating water pump was removed from service. This action did not sufficiently reduce the water level pressure differential across the screens and repeated attempts to restore the screens to service failed. At 0842 a normal reactor shutdown was commenced. The screens remained clogged with debris and unable to rotate.

As the screens bowed inward due to the high differential pressure, the flow appeared to begin to bypass the C screen at the outside vertical edges. The Superintendent of Power and the Operations Superintendent made a conservative decision to manually scram the reactor based on the degrading situation and the concern for continued availability of long-term cooling capability. At 0912 the reactor was manually scrammed from 45 percent power. Actions of the Abnormal Operating Procedure 1 for plant scrams were completed. Control rod 22-31 indicated "full out" on the full core display, but was verified to be full in by alternate indications. Control rod 30-07 inserted to notch position 02, the last notch before full in. This rod displayed the same anomaly during the last two scrams. The rod was manually inserted. The plant was stabilized. The reactor level and pressure control were established through use of the reactor feedwater pumps and the main steam bypass valves. At 0930 the A main condenser circulating water pump was removed from service to further reduce the differential pressure across the traveling screens. This action was effective in reducing the level differential across the screens to near normal conditions. As a conservative measure a plant cooldown was initiated.

Approximately 30 minutes after removal of the A main condenser circulation pump from service and consequent reduction in differential pressure across the screens, the A traveling screen was returned to service. The C screen was damaged (bowed) and could not be restored to service until repair parts were obtained. The plant cooldown was terminated at 1135 at approximately 500 psig reactor pressure. The B traveling screen (which was initially out of service for preventive maintenance) was restored to service at approximately 1500. The plant returned to service at 0418 on 10/21/90. The time off line was 43 hours, 6 minutes.

Cause

When the B screen had been removed from service, the input signal from the B traveling screen to the screen differential pressure indication system was not isolated. When the B intake bay was isolated and pumped down to service the screen, differential pressure indication across the screens was unintentionally lost and indicated downscale on instruments. With the differential indication and alarm disabled, the operating shift was not aware of the buildup of debris. More importantly, disabling of this signal also prevented the automatic high differential level actuation of the traveling screen system as they began to accumulate debris. Early automatic operation initiated by high differential pressure might have been adequate to prevent the excessive debris accumulation, subsequent excessively high differential pressure, and resulting shear pin failure and screen damage.

A review of the event found that the applicable operating and maintenance procedures failed to explicitly identify the need to isolate the differential pressure instrument system from a specific intake bay when it is pumped down for servicing the traveling screens.

A unique combination of wind conditions in which a strong generally southerly wind occurred at a time of near peak tree leaf-drop during dry weather resulted in movement of unusually large number of leaves from the land into the lake. During the 2300 to 0700 shift the wind direction reversed to come from a generally northerly direction driving the leaves back closer to shore. The increasing wind velocity created waves which submerged the leaves where they were drawn into the intake structure. As a random event, this set of circumstances occurred coincidentally with the prior removal of one of the three of the traveling screens from service for scheduled preventive maintenance.

The failure of the shear pins occurred due to the high differential loading on the screens which occurred following the windshift in the early morning hours. Operator observations of the system operation at 0200 and 0600 were based on observed screenwash flow to the debris collection basket. These activations are believed to have been initiated by the once every four-hour timer. Actual rotation of the screens was not verified. Because screenwash flow is only an indirect indicator of screen operation it is possible that, although the automatic timer initiated the screenwash, the screens themselves were not rotating due to earlier failure. The screens are enclosed. Removal of an inspection plate is required to actually confirm screen conditions. If, in fact, the screens were not rotating, then the screenwash water, although operating, would not have moved debris into the collection basket. Both inservice screens A and C had been inspected and lubricated (as part of the preventive maintenance program) within seven days prior to the event and found to be operating properly.

Analysis

The manual scram of the reactor is reportable under the provisions of 10 CFR 50.73 (a)(2)(iv) as an activation of an engineered safety feature actuation system [JE]. The plant systems responded as required to the manual scram. Appropriate isolations and trips were received during the low water level transient in the reactor vessel. The main condenser circulating water pumps operated continuously with no evidence of cavitation. Because these pumps have a higher suction water level requirement than the safety-related emergency service water and residual heat removal service water pumps, the continued availability of safety-related pumps was assured.

Corrective Actions

1. Procedural guidance has been developed or enhanced for:

   a. Removal of traveling screens from service.

   b. Operator response to high screen differential pressure

   c. Conditions when it is appropriate to place the traveling screens in manual continuous operation

2. The traveling screen differential pressure indication has been added to the operator round sheets.

3. The traveling screen differential pressure instrumentation will be:

   a. Added to the balance of plant instrument calibration schedule

   b. Added to the plant flow diagram drawings

Additional Inforamtion

Failed Component Data:

Plant Component Identification:	36TS-2A and 36TS-2C
NPRDS Component Code:	FILTER
Manufacturer:	Jeffrey Manufacturing Co.
NPRDS Vendor Code:	J033
Design Flow:	125,000 GPM
Maximum Differential Pressure:	70 PSI

WOLF CREEK
NUCLEAR OPERATING CORPORATION

John A. Bailey
Vice President
Nuclear Operations

November 23, 1990

NO 90-0291

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Subject: Docket No. 50-482: Licensee Event Report 90-023-00

Gentlemen:

The attached Licensee Event Report (LER) is being submitted pursuant to 10 CFR 50.73 (a) (2) (iv) concerning an Engineered Safety Features actuation.

Very truly yours,

John A. Bailey

John A. Bailey
Vice President
Nuclear Operations

JAB/jra

Attachment

cc: A. T. Howell (NRC), w/a
R. D. Martin (NRC), w/a
D. V. Pickett (NRC), w/a
M. E. Skow (NRC), w/a

P.O. Box 411 / Burlington, KS 66839 / Phone: (316) 364-8831
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U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Wolf Creek Generating Station	0 5 0 0 0 4 8 2	1 OF 0 3

TITLE (4): Engineered Safety Features Actuation - Switchyard Breaker Failure Causes Partial Loss Of Offsite Power and Shut Down Sequencer Actuation

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)
1 0	2 3	9 0	9 0	0 2 3	0 0	1 1	2 3	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 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 (AREA CODE)
Merlin G. Williams - Manager Plant Support	3 1 6 3 6 4 - 8 8 3 1

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
X	F K	B K R	G 0 8 0	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 October 23, 1990 at 0133 CDT, Switchyard Breaker 345-90 failed causing a loss of power to the east bus in the Switchyard, No. 7 transformer, transformer XNB01 and 4160 volt Engineered Safety Features bus NB01. The loss of power to NB01 resulted in an automatic starting of Emergency Diesel Generator "A", shedding of loads from NB01, repowering NB01 from the Emergency Diesel Generator, and sequencing key loads back onto the NB01 bus by the Shut Down Sequencer. As expected, the Turbine-Driven Auxiliary Feedwater Pump started, and Control Room Ventilation Isolation, Containment Purge Isolation, and Fuel Building Isolation Signals were generated. All operable Engineered Safety Features equipment responded properly to the actuation signals.

On October 24, 1990 at 1619 CDT, with the faulty breaker isolated, power to bus NB01 was re-established via the normal offsite power system configuration. The initial investigation has shown the possibility of a metallic particle creating an arc path between the "A" phase internal corona shield and the wall of the breaker's interrupting tank. The breaker's internal corona shield has been replaced and the breaker was placed back into service at 1420 CST on November 16, 1990. Additionally, a visual inspection was made on the breaker's other two interrupting tanks to examine for metallic particles and no particles were found.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Wolf Creek Generating Station	0 5 0 0 0 4 8 2	9 0	— 0 2 3 —	0 0	0 2 OF 0 3

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

INTRODUCTION

On October 23, 1990 at 0133 CDT, a loss of power to Engineered Safety Features (ESF) Transformer XNB01 [EB-XFMR] and 4160 volt bus NB01 [EB-BUS] occurred, causing an Emergency Diesel Generator [EK-DG] start and a Shut Down Sequencer [JE-STC] actuation. All operable ESF equipment responded properly to the actuation signals. This event is being reported per 10 CFR 50.73(a)(2)(iv) concerning unplanned actuations of ESF equipment.

DESCRIPTION OF EVENT

On October 23, 1990, the unit was operating in Mode 1, Power Operation, at approximately 100 percent rated thermal power. At 0133 CDT, Switchyard Breaker 345-90 [FK-BKR] failed causing a loss of power to the east bus in the Switchyard, No. 7 transformer, transformer XNB01 and bus NB01. The loss of power to NB01 resulted in an automatic starting of Emergency Diesel Generator "A", shedding of loads from NB01, repowering NB01 from the Emergency Diesel Generator, and sequencing key loads back onto the NB01 bus by the Shut Down Sequencer.

The loss of power to several radiation monitors [IL-MON], which sample the Containment Building [ND] atmosphere, Control Building [NA] air intake, and Fuel Building [ND] atmosphere, resulted in signals to actuate a Control Room Ventilation Isolation, Containment Purge Isolation, and Fuel Building Isolation.

The undervoltage condition on NB01 also initiated actuation of the Turbine-Driven Auxiliary Feedwater Pump (TDAFP) [BA-P] and a Steam Generator Blowdown and Sample Isolation Signal. Motor-Driven Auxiliary Feedwater Pump "A" [BA-P] and Centrifugal Charging Pump "A" [BQ-P] were automatically started by the Shut Down Sequencer. The Shut Down Sequencer also separated the Essential Service Water System (ESW) [BI] from the normal Service Water System [KG] and automatically started ESW Pump "A" [BI-P]. At 0133 CDT, the Control Room Operators started ESW Pump "B" to ensure availability of components served by the "B" train of the ESW System.

Following verification of proper functioning of ESF equipment, Motor-Driven Auxiliary Feedwater Pump "A" and Centrifugal Charging Pump "A" were secured and placed in "pull-to-lock" to prevent re-actuation at 0136 CDT. Entry was also made into the Technical Specification (T/S) Action Statements for inoperability of these pumps and for the loss of one off-site power source. At 0210 CDT the TDAFP was secured and by 0330 CDT the ventilation systems were restored to normal configuration.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Wolf Creek Generating Station	0 5 0 0 0 4 8 2	9 0	— 0 2 3 —	0 0	0 3 OF 0 3

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

At approximately 0350 CDT, XNB01 was re-energized via bus SL-7 and bus SL-8. At 0426 CDT, NB01 loads were transferred to XNB01 and Emergency Diesel Generator "A" was secured shortly thereafter. The T/S Action Statement for loss of one off-site power source was exited at that time.

Motor-Driven Auxiliary Feedwater Pump "A" and Centrifugal Charging Pump "A" were returned to "normal-after-stop" at approximately 0430 CDT. The T/S Action Statements for these pumps were exited at this time. At approximately 0510 CDT the ESW pumps were secured. At 0603 CDT, faulty breaker 345-90 was isolated and at 0615 CDT the east bus was re-energized.

On October 24, 1990 at 1619 CDT, with the faulty breaker isolated, transformer XNB01 and bus NB01 were transferred back to the No. 7 transformer, which is the normal system alignment. This transfer was made after the bus was energized for a period of time to confirm its integrity.

ROOT CAUSE AND CORRECTIVE ACTIONS

This event was caused by a failure of Switchyard Breaker 345-90. The initial investigation has shown the possibility of a metallic particle creating an arc path between the "A" phase internal corona shield and the breaker's interrupting tank. This metallic particle may have been dislodged or relocated the previous day when sulfur hexafluoride gas was added to the breaker's interrupting tank as an arc quenching medium. This gas has been added to a breaker in the past with no adverse affects. This was the first instance in which the gas was added to breaker 345-90 since shortly after the start of commercial operation (approximately five years).

The breaker's internal corona shield has been replaced and the breaker was placed back into service at 1420 CST on November 16, 1990. Additionally, a visual inspection was made on the breaker's other two interrupting tanks to examine for metallic particles and no particles were found. The breaker vendor has been consulted to provide technical assistance regarding this failure and was present for the visual inspection on the interrupting tanks. The failed Switchyard Breaker is a General Electric Company breaker, Model Number HVB-362-40KA-1.

ADDITIONAL INFORMATION

All ESF equipment performed as designed, and there was no damage to plant equipment or any release of radioactivity. Therefore, at no time did conditions develop that could have posed a threat to the health or safety of the public.

There have been no previous similar occurrences in which a Switchyard Breaker failure has caused an ESF actuation.