James A. FitzPatrick
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New York Power Authority

Michael J. Colomb
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March 21, 1996
JAFP-96-0128

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

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: LER-96-003

Plant Shutdown Due to Degraded Control Rod Scram Times and Manual Scram Due to Leak in the Main Turbine Electro-Hydraulic Control System

Dear Sir:

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv), "Any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)" and 10 CFR 50.73(a)(2)(i)(A), "The completion of any nuclear plant shutdown required by the plant's Technical Specifications".

There are no commitments contained in this report.

Questions concerning this report may be addressed to Mr. James Costedio at (315) 349-6358.

Very truly yours,

MICHAEL J. COLOMB

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cc: USNRC, Region I
USNRC Resident Inspector
INPO Records Center

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

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

LICENSEE EVENT REPORT (LER)

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

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TITLE (4)
Plant Shutdown Due to Degraded Control Rod Scram Times and Manual Scram Due to Leak in the Main Turbine Electro-Hydraulic Control System

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
02	22	96	96	003	00	03	21	96	NA	05000
									NA	05000

OPERATING MODE (9): N
POWER LEVEL (10): 007

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

	20.2201(b)		20.2203(a)(2)(v)	X	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)	X	50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		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)
Mr. James Costedio, Licensing Engineer	(315) 349-6358

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	TG	TBG		N					

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

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

On 02/22/96 at 0333 hours, with the plant operating at 60 percent of rated power, results from control rod scram time testing revealed an increased scram time to Notch Position 46 for 19 out of 20 control rods tested. Based on an evaluation of the test results, a normal reactor shutdown was commenced. On 02/22/96, during the controlled shutdown, at 1406 hours, with the plant operating at approximately seven percent of rated power in the Startup/Hot Standby mode, a manual scram was initiated. The manual scram was initiated in anticipation of loss of main turbine bypass valve control due to an Electro-Hydraulic Control (EHC) system leak. The EHC leak was caused by damaged tubing. Results of an engineering evaluation indicate that the subject tube had been damaged due to fatigue. The stainless steel Fluid Actuator Supply (FAS) tubing to all four Turbine Bypass Valves was replaced. The primary cause of the delayed scram time response is attributed to the adherence of the Scram Solenoid Pilot Valves (SSPV) exhaust diaphragms to the valve seats. The 03SOV-118 SSPV exhaust diaphragms for all control rods were replaced.

NRC FORM 366A
(4-95)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

EIIS Codes are in []

EVENT DESCRIPTION

On February 22, 1996 at 0333 hours, with the plant operating at 60 percent of rated power, results from scram time testing revealed that the average scram time for 20 control rods [AA] to Notch Position 46 was greater than 0.338 seconds. Averaging in these 20 control rod scram times with previously measured control rod scram times for the remaining 117 control rods resulted in an average control rod scram time of less than the Technical Specification required 0.338 seconds. However, it was assume that remaining control rod scram times would increase if tested. Based on this, Technical Specification Action statement 3.3.E was entered which requires an orderly reactor shutdown be initiated and the reactor be in a cold condition within 24 hours.

On February 22, 1996, during the controlled shutdown, at 1406 hours, with the plant operating at approximately seven percent of rated power in the Startup/Hot Standby mode, a manual scram was initiated. Prior to the manual scram, loads had been transferred to the reserve source, the main generator [TB] had been removed from service, and the Main Turbine [TA] Bypass Valves (TBVs) were being used for pressure control. As TBV Number 2 closed, TBV Number 1 began to oscillate. The oscillations were approximately 10 to 20 percent of the valve's full stroke. The Electro-Hydraulic Control (EHC) [JG] Pump B discharge strainer high differential pressure annunciator alarmed intermittently. Recognizing abnormal system operation, the Shift Manager dispatched an operator to investigate. Subsequently, the operator reported an unusual noise in the area of the EHC pumps. EHC Pump A was placed in service and the high differential pressure alarm cleared. However, TBV oscillations continued. The EHC System Engineer and an Instrumentation and Control (I&C) supervisor were called to the control room [NA] to observe the oscillations. With TBV Number 1 still oscillating, the control room received a report of a water leak in the condenser [SG] area. A Non-Licensed Operator (NLO) was dispatched to the scene and reported a EHC fluid leak to the control room. The Shift Manager briefed the control room operators on the EHC leak, made operator panel assignments, and directed insertion of a manual scram. The leak was found on the tubing supplying a TBV.

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

The sequence of events leading up to and immediately following the manual scram is presented below:

February 22, 1996

03:33	Due to an increasing trend in control rod scram times, Technical Specification Action statement 3.3.E is entered, which requires an orderly reactor shutdown be initiated and the reactor be in a cold condition within 24 hours.
05:10	Transferred plant auxiliary electrical loads to the reserve source.
09:59	Removed the main generator from service.
13:23:20	TBV Number 1 oscillations begin, control rod insertions in progress for reactor shutdown. EHC pump B discharge strainer high differential pressure annunciator alarms.
13:24 (approx.)	Shift Manager (SM) dispatches Non-Licensed Operator (NLO) to investigate. Shift Manager notifies I&C supervisor and system engineer of EHC problems.
13:25 (approx.)	NLO reports to control room a high pitched noise in the area of the EHC pumps. EHC pump A is placed in service and the EHC pump B discharge strainer high differential pressure alarm clears.
14:00 (approx.)	Control room receives a report of a water leak in the condenser bay. SM dispatches NLO to the condenser bay. SM performs control room brief on potential EHC leak.
14:05	NLO reports to control room a EHC leak in the condenser bay. Radwaste operator is notified of EHC leak into turbine building floor drains [TF]. Radwaste operator secures all turbine building floor sump pumps.
14:05	SM assigns operators to Reactor Pressure Vessel (RPV) pressure and level control, and then SM directs insertion of manual reactor scram.
14:06:10	Control room operator inserts manual reactor scram.

14:06:30 (approx.)	Average Power Range Monitors (APRMs) [IG] go downscale and the mode switch is placed in shutdown. The feedwater control system [JB] startup flow control valve responds to automatically control RPV level.
14:07	Control room operator secures the EHC system.
14:07:40	All control rods verified full in.
14:10:17	Reactor scram reset.
14:12:15	Control Room Operator lowers turbine seal steam [TC] pressure to control plant cooldown.
14:13	Residual Heat Removal (RHR) [BO] Loop A placed in full torus cooling in anticipation of Safety Relief Valve (SRV) [SB] operation for reactor pressure control.
14:14:54	Torus water level drops to EOP-4 (Primary Containment Control) entry level.
14:18	Commenced adding water to the torus via RHR Loop A.
14:19 (approx.)	Control Room Operator trips recombiner [WF] to control plant cooldown.
14:40	Exited EOP-4. Secured making up to the torus via RHR Loop A.
18:44	Started Reactor Core Isolation Cooling (RCIC) [BN] in pressure control mode to maintain cooldown rate.
19:48	Started High Pressure Coolant Injection (HPCI) [BJ] in pressure control mode to maintain cooldown rate.
19:51	Tripped RCIC to control cooldown rate.
20:09	HPCI low pressure isolation at RPV pressure of 77.9 psig.
20:20	Started RCIC in pressure control mode.
20:39	RCIC low pressure isolation at RPV pressure of 76.0 psig.

21:28 Started Reactor Feed Pump (RFP) [SJ] B to continue cooldown.

22:39 Opened SRV E to continue cooldown.

22:42 SRV E closed on low RPV pressure at 48 PSIG.

23:54 Opened RFP A steam drains to maintain cooldown.

February 23, 1996

00:24 Restored EHC system to service. A temporary modification had previously been installed to repair the cracked tubing.

00:37 Continued cooldown using manual operation of TBV.

01:45 Reactor vessel temperature less than 212 degrees Fahrenheit (entered the cold condition).

CAUSE OF EVENT

EHC Tubing Leak

Investigations revealed a cracked 1-1/4 inch diameter flared stainless steel tube section from TBV Number 2.

Visual examination revealed two circumferential oriented through-wall cracks, located diametrically opposite from each other, adjacent to the flared end (beneath the compression nut) of the tube. The cracks arrested prior to merging and thus prevented complete severing of the tube. In addition, the tube had permanently deformed in the vicinity of the cracks, such that the tube had bent about an axis parallel to the crack planes and perpendicular to the tube axis. The occurrence of diametrically oriented cracks, combined with the observed deformation, indicates that the tube had been subjected to reversed bending.

Visual and metallographic examinations, as well as scanning electron microscopy (SEM), revealed that cracking had initiated at the outer surface, and propagated through the wall thickness and about the circumference. The crack profiles were similar in appearance, exhibiting relatively straight and triangular profiles. This is characteristic of progressive cracking in the nature of fatigue. SEM examination revealed the presence of fatigue striations which indicate that the tube had been subjected to fatigue loading.

Results of the engineering evaluation indicate that the subject tube had been damaged due to fatigue [Cause Code B]. This fatigue may be due to piping vibration caused by vibration at the bypass valve or EHC system pressure fluctuations.

Degraded Control Rod Scram Times

A General Electric (GE) safety assessment reveals that the primary cause of the delayed scram time response is due to the adherence of the solenoid valves exhaust diaphragms to the valve seats. The underlying mechanism may be related to the physical chemistry of the Viton polymer and its surface interaction with the brass seat. Apparently, the adherence characteristic is sufficient to cause a slight hesitation of the valves exhaust diaphragms to separate from the valve seats, but not to a degree which results in a complete failure of the valves to operate.

ANALYSIS

Manual Scram

This event is reportable under 10 CFR 50.73 (a)(2)(iv), which requires licensees to report "Any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)".

This event is bounded by the previously analyzed turbine trip from low power without bypass as described in the FitzPatrick Updated Final Safety Analysis Report (UFSAR). The plant responded as designed following the manual scram from approximately seven percent of rated power without bypass valve control. There was no challenge to the reactor coolant pressure boundary or the fuel cladding integrity. Therefore, the safety significance of this event was minimal.

The Post Transient Review revealed that the Shift Manager prepared for and directed insertion of a manual reactor scram when faced with an imminent loss of the EHC system. The operating crew took manual actions to control RPV and primary containment parameters within prescribed limits, and to meet a Technical Specification 24 hour cold condition action statement.

Degraded Control Rod Scram Times

This event is reportable under 10 CFR 50.73(a)(2)(i)(A), which requires licensees to report "The completion of any nuclear plant shutdown required by the plant's Technical Specifications".

GE has performed an interim safety assessment of the Viton dual-type SSPV response time delay. This assessment states that while reactor scram is an important BWR safety function, small degradations of the 5 percent (approximately Notch Position 46) insertion scram time does not have a significant impact on the consequences of design basis accidents and applicable anticipated operational occurrences.

GE has periodically evaluated the impact of slower scram speeds for either individual, several, or all control rods. These evaluations have shown that scram speed is only of importance for the rapid transients, such as rapid pressurization events. These evaluations confirm that the important parameter affecting the consequences of most safety analyses is the integrated power, rather than the peak power. As such, scram speed has little influence on the integrated power for many events such as loss of feedwater heating, and core and containment cooling following a postulated Loss of Coolant Accident (LOCA).

For rapid pressurization events, the slower scram speed can influence the peak power, and consequently, the fuel thermal limits. GE fuel licensing methodology for calculating the Minimum Critical Power Ratio (MCPR) Operating Limit (OL) includes assumptions regarding control rod scram speed, and slower scram speeds may require larger MCPR OLs. Although not specific for FitzPatrick, GE evaluations supporting Improved Standard Technical Specifications (NUREG-1433) demonstrate that increasing the 5 percent insertion time limit from the current value of 0.375 seconds to 0.490 seconds, requires raising the MCPR OL less than 0.01. Because the observed average time to Notch Position 46 corresponds to a 5 percent insertion time less than 0.490 seconds, the impact of degraded scram time performance on plant transient response is negligible.
The average scram time to Notch Position 46 for the 20 control rods tested on February 22, 1996 was 0.373 seconds.

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An additional conservatism in analyzed transient response is the assumption that all rods scram at the average speed. In reality there is a distribution of rod scram speeds about the average, with faster moving rods having more of an affect on transient response than slower moving rods (i.e., the net affect of a distribution in scram speeds would be to improve transient response).

Based on the above considerations, the impact of the slower scram speed to Notch Position 46 (Approximately 5 percent insertion) did not present a significant impact on safety.

GE and the Authority as a member of the Boiling Water Reactor Owners Group (BWROG) are working on a final long term solution.

CORRECTIVE ACTIONS

1. Fluid Actuator Supply (FAS) tubing to all four TBVs was replaced with flexible hoses prior to plant startup.

2. A walkdown of accessible EHC piping and supports for any evidence of degradation was conducted prior to plant startup.

3. Fluid Jet Supply (FJS) tubing to TBVs 1 and 2, that were found damaged during the above noted walkdown, were replaced.

4. Temporary monitoring equipment was installed and additional data was obtained during plant startup in order to evaluate vibration at the TBVs and EHC system pressure fluctuations.

5. The need for any additional corrective actions necessary to eliminate vibration at the TBVs will be evaluated. **DUE DATE: JUNE 30, 1996**

6. The need for any additional corrective actions necessary to eliminate EHC system pressure fluctuations will be evaluated.
DUE DATE: JUNE 30, 1996

7. The 03SOV-118 SSPV exhaust diaphragms for all control rods were replaced.

8. Scram time testing for all control rods was completed satisfactorily following the replacement of the 03SOV-118 SSPV exhaust diaphragms.

9. Accelerated scram time testing will continue to be conducted to monitor the performance of the Viton diaphragms.

ADDITIONAL INFORMATION

Previous Similar Events: No previous events at FitzPatrick involving EHC system leaks have resulted in automatic or manual reactor scram.

Failed Components: EHC system 1-1/4 inch diameter stainless steel tube section near the flared end.