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GGC-94-057

March 25, 1994

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Reference: Quad Cities Nuclear Power Station
Docket Number 50-254, DPR-29, Unit One

Enclosed is Licensee Event Report (LER) 94-004, Revision 00, for Quad Cities Nuclear Power Plant Station.

This report is submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(ii)(b). Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers being seriously degraded or that resulted in the nuclear plant being in condition that was outside the design basis of the plant.

The following commitments are being made by this letter:

1. After management review and approval, the FW flow instrumentation shall be recalibrated to the calculated flow coefficients based on finalized test data.

2. Station Management shall determine if further inspection and testing of the FW flow nozzles is needed.

3. GE SIL No. 452, Revision 1, shall be reviewed by the station, and corrective actions implemented if deemed necessary.

4. A supplemental report to this LER will be submitted to inform the NRC of the stations' actions to address GE SIL No. 452, Revision 1 concerns.

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If there are any questions or comments concerning this letter, please refer them to Nick Chrissotimos, Regulatory Assurance Administrator at 309-654-2241, ext. 3100.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD CITIES NUCLEAR POWER STATION

G. G. Campbell
Station Manager

GGC/TB/plm

Enclosure

cc: J. Schrage
C. Miller
INPO Records Center
NRC Region III

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LICENSEE EVENT REPORT (LER)

Form Rev. 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Quad Cities Unit One	0 5 0 0 0 2 5 4	1 of 0 6

Title (4)

Results Of Feedwater Flow Testing Found Flow Indication In The Non-conservative Direction.

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)
									Quad Cities Unit 2	0 5 0 0 0 2 6 5
0 2	2 8	9 4	9 4	-- 0 0 4 --	0 0	0 3	2 5	9 4		0 5 0 0 0

OPERATING MODE (9): 04

POWER LEVEL (10): 0 9 2

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

20.402(b)	20.405(c)	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)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	X 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	
Jon Brigl, Regulatory Assurance Ext. 3115	3 0 9	6 5 4 - 2 2 4 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	S J	N Z L	P 1 7 5	N					

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 1	0 1	9 6

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

ABSTRACT:

In February, 1994, Special Test 1-176, Rev. 1, was performed to determine the difference in actual versus measured Feedwater (FW) flow for both units at Quad Cities Station. The results of the tests indicated that measured FW flow could be non-conservatively low by as much as 1.40% (Unit-1) and 1.70% (Unit 2). The measurement error in FW flow indicates that past operation of the units at full power exceeded the licensed steady state power level of 2511 Megawatts thermal by 1.56% (Unit-1) and 1.78% (Unit-2).

The Causal Factors for this event are attributed to <u>Design Configuration and Analysis</u>, and <u>Equipment Specification, Manufacturer and Construction</u>.

Corrective actions taken prior to Special Test 1-176, Rev. 1, was to limit thermal power of both reactors to 97% of rated thermal power, and the Average Power Range Monitors and Rod Block Monitors setpoints were setdown by 1%. Additional corrective actions are to recalibrate the FW flow instrumentation to calculated flow coefficients, determine if further inspection and testing of the FW nozzles is needed, and to address concerns associated with GE SIL NO. 452, Revision 1. A supplemental report will be submitted to inform the NRC of actions taken to address GE SIL No. 452, Revision 1 concerns.

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PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2511 MWt rated core thermal power.

EVENT IDENTIFICATION: Results of Feedwater Flow Testing found flow indication in the non-conservative direction.

A. CONDITIONS PRIOR TO EVENT:

Unit: One	Event Date: February 28, 1994	Event Time: N/A
Reactor Mode: 4	Mode Name: RUN	Power Level: 92%

This report was initiated by Licensee Event 254\94-004.

RUN (4) - In this position the reactor system pressure is at or above 825 psig, and the reactor protection system is energized, with APRM protection and RBM interlocks in service (excluding the 15% high flux scram).

B. DESCRIPTION OF EVENTS:

On September 23, 1993, the NRC Diagnostic Evaluation Team (DET) expressed concerns associated with reactor feedwater (FW) [SJ] flow measurement. The NRC DET expressed concern, involving possible FW flow nozzle [NZL] measurement uncertainties, after reviewing the Vulnerability Assessment Team (VAT) report (performed in the fall of 1992). The VAT report was generated by CECo personnel.

Per request from the Operations Department, Nuclear Fuel Services (NFS) performed an Operability Evaluation (NFS:BND: 93-092) on September 24, 1993. The evaluation concluded the FW flow nozzles were operable, and recommended a 1% derating of the core thermal power. Consequently, the Operations Department issued special instructions to limit thermal power, of both reactors, to 97% of rated thermal power. This action was immediately taken to ensure both units remained below rated thermal power until the FW flow concern could be resolved. Within 48 hours the Average Power Range Monitors (APRM) and Rod Block Monitors (RBM) setpoints were setdown by 1%. These actions were utilized to ensure that the fuel cladding integrity safety limit, and Minimum Critical Power Ratio (MCPR) were met.

On February 7 (Unit-1) and February 21 (Unit-2), 1994, Special Test 1-176, Rev.1, was performed to determine the difference in actual versus measured FW flow. The results of the test, received February 14 (Unit-1) and 28 (Unit 2), 1994, indicated that measured FW flow could be non-conservatively low by as much as 1.40% (Unit-1) and 1.70% (Unit-2). During Special Test 1-176, Rev.1, the power level of Unit-1 was 92 % power, and Unit-2 was 95% power.

The error in measurement of FW flow indicates that past operation of both units at calculated full power exceeded the licensed steady state power level of 2511 Megawatts thermal (MWt) by 1.56% (Unit-1) and 1.78% (Unit-2).

Problem Identification Form (PIF) 94-0006 was generated by the System Engineer to investigate this condition, evaluate safe continued plant operation, and report this event as an LER exceeding the stations Technical Specifications.

C. <u>CAUSE OF THE EVENT:</u>

The exact cause of the 1.40% (Unit-1) and 1.70% (Unit-2) non-conservative measurement of FW flow could not be determined. Several possibilities were investigated, and are included below. The most probable cause of this condition is a combination of the causes in varying degree.

Possible erosion and damage of the FW nozzles. Industry experience indicated that other plants have experienced non-conservative errors in FW flow measurement. The errors have been attributed to erosion which changed the flow element geometry such that indicated flow was in the non-conservative direction (see Section F for more information).

An internal inspection of both units FW nozzles was performed prior to Special Test 1-176, Rev. 1. The inspection used a boroscope, and access was gained through the nozzles inspection ports. No abnormalities dealing with erosion or damage were noted in the nozzles or sensing lines.

Plant FW flow instrumentation inaccuracy and calibration techniques. General Electric (GE) SIL No. 452, supplement 1, was issued in 1988 that raised concern with respect to the uncertainty associated with the FW flow instrumentation. Recommended corrective actions from SIL No. 452 were addressed early in 1989 by the Instrument Maintenance Department (NTS 254-455-88-45201S1), with respect to instrumentation full span pressure drop calculation adjustments, flow transmitter calibration adjustments and the process computer flow measurement accuracy was checked.

A Revision 1 to SIL No. 452, issued February 16, 1994 to the nuclear industry, involves additional FW flow nozzle transmitter calibration concerns. The station is investigating the SIL revision, and will implement corrective actions if deemed necessary.

The following is a summary of conclusions and Causal Factors (C/F) which may have contributed to equipment malfunctions.

C/F: Design Configuration and Analysis

The FW flow nozzles were originally installed with less reliable means of measuring the flow coefficient. The original FW flow nozzles were built to specified tolerances by GE, and welded in place. Sodium and lithium tracer tests have been used at other stations, for calibrating nozzles that are welded construction, with inconsistent results. Technology has advanced to where a more reliable test now exists (rubidium nitrate tracer test), and Quad Cities Station is the second nuclear station known to use it for FW flow measurement.

CF: Equipment Specification, Manufacture and Construction

The FW flow nozzle discharge coefficient may have been inadequately applied in 1974. Originally, GE provided a FW flow nozzle manufactured to a specified tolerance of 1%. Modifications M4-1(2)-74-012 moved the FW nozzle throat tap downstream to eliminate potential errors associated with bypass leakage. This modification consisted of changing the discharge coefficient of the FW nozzles. Based on nozzle calibration data from Browns Ferry Unit-2, which were modified in the same manner, GE assigned a discharge coefficient to FW nozzles at Quad Cities station. The FW nozzles at Quad Cities were not physically calibrated after the modification changed the nozzle configuration.

Because the inaccuracy of FW flow determination did not create a significant safety concern, and due to self identification by the manufacturer through industry SIL's, this event is not 10CFR21 reportable.

D. SAFETY ANALYSIS

The safety significance of this event is minimal. Review of past and present transient and accident analysis methodology determined that an overpower condition of approximately 1.56% (Unit 1) and 1.78% (Unit-2) would not have exceeded the 2% overpower initial condition assumed in the analysis. Therefore, no safety limit or fission product boundary would have been compromised during normal, abnormal or accident conditions, due to this nonconservative measurement of FW flow.

The measured FW flow uncertainty is applied in the statistical analysis utilized by GE in the fuel bundle design document NEDE-24011(P)(A) to establish the fuel cladding integrity Safety Limit Minimum Critical Power Ratio (SLMCPR) value. The FW flow uncertainty is then combined with other uncertainties to establish confidence that 99.9% of the fuel rods do not experience a Departure from Nucleate Boiling (DNB) during a transient. The analytical uncertainty is used in FW flow is 1.76%, which bounds the measured FW flow error of 1.40% (Unit-1) and 1.70% (Unit-2). Therefore, operating with the measured FW flow error did not reduce the required margin to the SLMCPR. The 1.76% or greater uncertainty was applied throughout the operating history of both Units-1 and 2. The uncertainty values are tabulated in GE's thermal hydraulic analysis document NEDE-31152P.

The accident analysis for a Loss of Coolant Accident for both of Quad Cities units assume core thermal power conditions which are 102% of rated thermal power. Hence, throughout the operating history of Unit-1 and 2, the fuel integrity limit following a loss of coolant accident would not be affected by an increase in core thermal power of 1.56% and 1.78%, since the calculations were always performed at 102% of rated core thermal power.

The Linear Heat Generation Ratio (LHGR) limit is established to prevent fuel clad cracking due to differential expansion of the fuel pellet. It is based on the peak fuel pin power level which would result in a 1 % plastic strain deformation of the clad. The conservatism provided in the LHGR limit for GE fuel bounds the 1.56% and 1.78% power increase due to the FW flow uncertainty. A review of the operating and failed fuel history for Unit-1 and 2 indicates that adequate margin existed to the LHGR limit throughout the history of the units.

In general the non-conservative FW data results reveal that the fuel cladding integrity safety limit was never compromised by the affect of 1.40% and 1.70% non-conservatism because it was enveloped by the analytical uncertainty of 1.76% throughout the operating history of Unit-1 and 2. The previous adjustments (3% derate, 1% setdown of APRM setpoints) at Quad Cities Units are considered conservative until the new FW discharge coefficients, for both Quad Cities units, are provided by GE.

E. <u>CORRECTIVE ACTIONS:</u>

Based on NFS Operability Evaluation (NFS:BND:93-092) the immediate corrective action by Quad Cities Station management was to issue special instructions that reactor thermal power for Unit-1 and Unit-2 shall not exceed 97% (from 2511 MWt to 2435 MWt). Additionally, the APRM's and RBM setpoints were setdown 1 % to ensure adequate margin to Technical Specifications and Fuel Thermal Limits.

Remaining corrective actions include:

1. After management review and approval, the FW flow instrumentation shall be recalibrated to the calculated flow coefficients based on finalized test data. (NTS# 2541809400401).

2. Station Management shall determine if further inspection and testing of the FW flow nozzles is needed. (NTS# 2541809400402).

3. GE SIL No. 452, Revision 1, shall be reviewed by the station, and corrective actions implemented if deemed necessary. (NTS# 2541809400403).

4. A supplemental report to this LER will be submitted to inform the NRC of the stations' actions to address GE SIL No. 452, Revision 1 concerns. (NTS# 2541809400404).

F. PREVIOUS OCCURRENCE:

A nationwide Nuclear Plant Reliability Data System (NPRDS) search was performed, and no failures were found involving FW flow nozzles.

Industry experience shows that Calvert Cliffs, Oyster Creek, Callaway and Brunswick stations have experienced non-conservative errors in FW flow measurement. Calvert Cliffs, Callaway and Brunswick attributed their errors to erosion of the carbon steel piping around the high pressure taps or the nozzles. The erosion changed the flow element geometry such that indicated flow was in the non-conservative direction. Oyster Creek could not determine the exact cause. Existing plant instrumentation accuracy, calibration techniques, a change in nozzle flow coefficient or FW piping erosion are considered contributors to the flow difference observed.

After review of the Nuclear Tracking System data base, there were no LER's at Quad Cities Station involving FW flow nozzles.

G. COMPONENT FAILURE DATA:

There was no component failure associated with this event.

The FW nozzles are manufactured by Permutit Co., Dwg # 528-50630. Made to GE Specification #21A5614.