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LWP 95-111

ComEd

December 5, 1995

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

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

Subject: Licensee Event Report (LER) 254/94-004 Supplemental Information.

As stated in LER 254/94-004, supplemental information is being provided and is enclosed as Attachment 1. This information constitutes revision 01 to the original LER documentation.

Attachment 2 is a reproduction of the original text of LER 254/94-004.

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 a condition that was outside the design basis of the plant."

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 STATION

L.W. Pearce

L.W. Pearce
Station Manager

Attachment 1- LER Supplemental Information
Attachment 2- LER 254/94-004 (Copy)

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

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Quad Cities Nuclear Power Station System Engineering Transmittal

To:	**L. W. Pearce**	**Prepared By:**	**R. E. Dvorak** RED
From:	**D. Craddick** DC	**Reviewed By:**	**S. Davis** SMD
Date:	**12/18/95**		

ATTACHMENT 1

This supplemental LER report is being submitted to inform the Nuclear Regulatory Commission of Quad Cities Stations' actions to address GE SIL 452 Supplemental 1 Revision 1 in accordance with LER 94-004. GE SIL 452 Supplemental 1, Revision 1 addressed concerns with the calibration of feedwater flow element transmitters.

The following actions have been taken in accordance with the recommendations of SIL 452 Supplemental 1, Rev 1:

Recommendation 1:

Inspect the feedwater flow nozzles at periodic intervals to maintain the calibration basis. On-line methods of discharge coefficient verification (ie: ultrasonics, tracer tests) may be utilized as a guide for inspection frequency.

> The feedwater flow nozzles were calibrated in 1994 utilizing tracer methodology under Tests 1-176, 1-178, 2-110, and 2-112. The nozzles were not individually calibrated prior to original installation. The feedwater flow nozzles are inspected every other outage via an inspection port just upstream of the nozzle to maintain this tracer calibration. This activity is maintained within the station surveillance tracking (GSRV) program to ensure its frequency of completion. In addition, the station has begun testing on the measurement of feedwater flow utilizing ultrasonic techniques. This method of feedwater flow measurement has not been permanently installed.

Recommendation 2:

Include calibration of the feedwater flow transmitters in instrument maintenance planning for each refuel outage.

> The feedwater flow transmitters are calibrated once every refuel outage in accordance with the GSRV program.

Recommendation 3:

Review procedures for calibrating feedwater flow transmitters to eliminate potential biases. Also, review the full scale differential pressure calculation for potential errors.

A procedure for calibrating the feedwater flow transmitters has been issued effective 12/26/95. The procedure addresses zero and span shifts to ensure these biases do not affect flow measurement. The problem associated with transmitter calibration was also acknowledged in the response to NRC NOV 94-005.

The resistors used for deriving the feedwater flow from the output of the transmitters were verified to be accurate in Nuclear Work Requests Q15256 and Q15257.

The full span pressure drop calculation was reviewed for correctness and applied to the results of Tests 1-176, 1-178, 2-110, and 2-112 for the new nozzle coefficient calculation. The correct thermal expansion factor was being used in the calculation. An error was found in the original span calculation which was corrected in the subsequent calculation, documented with Problem Identification Form 94-039, and placed on the Nuclear Network.

Recommendation 4:

Determine whether the appropriate stainless steel area factor was included in the feedwater temperature compensation process computer calculation.

The process computer feedwater temperature compensation calculation was reviewed with General Electric assistance and found to be acceptable.

Recommendation 5:

Verify that the measured feedwater temperature is corrected for any factor which may change the temperature from which is measured at the nozzle.

The core thermal power affect as described in the SIL from the Reactor Water Cleanup system does not apply due to the Quad Cities plant configuration.

Stratification of the final feedwater temperature has been identified as a factor affecting the process computer feedwater flow temperature correction. This affect was documented in PIF 95-0111 and may occur only when one of the three final feedwater heaters is removed from service. A temperature stratification may occur which causes the final feedwater temperature to be erroneously indicated. The affect on feedwater flow and core thermal power was evaluated in PIF 95-0111 response and was found to be bounded by the existing uncertainty analysis. Further, plant procedures require the unit to be derated when any final feedwater heater is removed form service.

Thus, factors which affect the feedwater temperature have been evaluated and determined not to have a detrimental affect on the feedwater flow measurement.

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number: LER 1-94-004 Supplement 1	Date: 12/22/95
Subject: Results of Feedwater flow testing found flow indication in the non conservative direction	
Submitted by: Terry Barber	

FOR REVIEW:

1. Safety Evaluations NOT involving an unreviewed safety question as defined in 10CFR50.59 for:
   - [ ] a. Changes to procedures as described in the Safety Analysis Report.
   - [ ] b. Changes to equipment or systems as described in the Safety Analysis Report.
   - [ ] c. Tests or experiments NOT described in the Safety Analysis Report.
2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
   - [ ] a. Procedure changes.
   - [ ] b. Equipment or system changes.
   - [ ] c. Tests or experiments.

- [ ] 3. Proposed changes to the Technical Specifications or Operating License.
- [ ] 4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
- [ ] 5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
- [x] 6. All REPORTABLE EVENTS (LERs only).
- [ ] 7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
- [ ] 8. All changes to the Station Emergency Plan prior to implementation.
- [ ] 9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

- [ ] 10. Other OSR Items/Documents NOT addressed above.

This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) for information only. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.