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September 9, 1996

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Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

SUBJECT: Byron Station Unit 1 Steam Generator Interim Plugging Criteria 90 Day Report for the End-of-Cycle 7 Inspection

Byron Nuclear Power Station
Facility Operating License NPF-37
NRC Docket No. 50-454

References:

1. November 9, 1995, Letter from M.D. Lynch (NRR) to D.L. Farrar (ComEd) Issuing Amendment No. 77 to Facility Operating Licenses NPF-37 and 66, Docket Nos. STN 50-454 and STN 50-455.

2. NRC Generic Letter 95-05, "Voltage-Based Criteria for the Repair of Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," dated August 3, 1995.

3. September 20, 1995, Letter from H. Pontious (ComEd) to NRC regarding the September 13, 1995, Teleconference between ComEd and NRC concerning the increase in Interim Plugging Criteria.

4. March 19, 1996, Letter from D. Saccomando to NRC regarding ComEd implementation of 3.0 volt Interim Plugging Criteria Probe Wear Criteria.

In Reference 1, NRC approved a license amendment for Byron Station to implement a voltage-based Interim Plugging Criteria (IPC) for Unit 1 through Cycle 8 for Outer Diameter Stress Corrosion Cracking (ODSCC) in steam generator tubing. NRC Generic Letter 95-05 (Reference 2) served as guidance for this amendment. NRC Generic Letter

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95-05 requires that the final results of the steam generator inspection and tubing integrity evaluation be submitted to the Staff within 90 days of plant restart following a steam generator inspection that implemented a voltage-based repair criteria. Reference 3 requires a complete report to be submitted to the Staff within 90 days of plant restart detailing the results of steam generator internal visual inspections performed to support the increased IPC. In addition, Reference 4 requires the IPC 90 day report that is submitting to the Staff address the implementation of the alternate probe wear criteria used during the eddy current inspection.

Pursuant to these reporting requirements, Commonwealth Edison (ComEd) is submitting the enclosed reports concerning the end-of-cycle 7 refuel outage steam generator IPC inspections and tube integrity evaluation. Attachment A to this letter contains the final report of the steam generator internal visual inspections performed to verify integrity of components necessary to support 3.0 volt IPC. Attachment B to this letter contains the results of the steam generator IPC eddy current inspection, results of the tube integrity evaluation, and assessment of the alternate probe wear criteria implementation.

Please address any questions regarding this matter to Ms. Marcia Lesniak, Byron Station Nuclear Licensing Administrator at Downers Grove 708/663-6484.

Sincerely,

David B Wozniak

David B. Wozniak
Site Engineering Manager
Byron Nuclear Power Station

DBW/JS/cb

Attachments

cc: A. B. Beach, Regional Administrator - RIII
S. Burgess, Senior Resident Inspector - Byron
G. Dick, Byron Project Manager - NRR
Office of Nuclear Facility Safety - IDNS

ATTACHMENT A

Tube Support Plate Integrity Verifications
Byron Unit 1 Cycle 7 Refuel Outage (B1R07)

The structural integrity of steam generator internal components that are important to the bases of 3.0 volt IPC were inspected during the Byron Unit 1 mid-cycle inspection in the Fall of 1995 in accordance with the "SG Structural Integrity Plan in Support of Braidwood-1 and Byron-1 3.0 volt IPC" (Inspection Plan). Additionally, Byron committed to perform additional structural load path inspections during B1R07, as described in a September 8, 1995, letter to the Staff from K.L. Graesser (Byron Letter 95-0308). The additional inspections included visual inspection of the vertical support bar welds below the flow distribution baffle, verification of the tube bundle wrapper alignment, enhanced eddy current of tubes near the anti-rotational device, and eddy current verification of tube support plate presence.

A visual inspection of the vertical support bar welds (24) beneath the flow distribution baffle was performed in each steam generator. The inspection was performed following completion of sludge lancing operations. Proper lighting and resolution was verified to meet ASME VT-1 requirements. Degradation of the welds was not found in any location.

Tube bundle wrapper alignment in all four steam generators was verified through each of the four sludge lance inspection ports located 90 degrees apart. Each wrapper was visually verified to be aligned at the four inspection ports just above the tubesheet. The ability to install the sludge lancing equipment also ensured no misalignment between the steam generator shell and the tube bundle wrapper.

Enhanced eddy current examinations were performed in the areas of the three anti-rotation devices in each SG using the EPRI developed technique. The focus of this inspection was to verify the integrity of the tube support plate. The enhanced technique involved acquiring data with a bobbin coil probe at a reduced pull speed of 12 inches per second or less. Anomalies, if found, were to be compared to defect signals from laboratory support plates fabricated and tested by EPRI. Fifty (50) intersections were inspected at each anti-rotational device. Due to SG symmetry, 75 tubes were inspected to encompass the 50 intersections per anti-rotation device. Data was collected for the entire tube and each support plate was evaluated. No anomalies indicative of degradation were detected in any SG.

The presence of each tube support plate was verified for all SG tubes. This was performed as part of the normal eddy current analysis of each tube.

ATTACHMENT B

Steam Generator Tube Integrity Evaluation

Westinghouse Report SG-96-08-005

Byron Unit-1 End-of-Cycle 7B
Interim Plugging Criteria Report
August 1996