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Subject: Additional Information Regarding an Axial Crack Found in Steam Generator Tube 58-119 During 10RFO

Ladies and Gentlemen:

The purpose of this letter is to provide a summary of Toledo Edison's (TE) findings with regard to the portion of Steam Generator tube number 58-119 removed from Davis-Besse Nuclear Power Station's (DBNPS) Steam Generator A during the Tenth Refueling Outage (10RFO). This summary was previously requested verbally by the NRC Staff for informational purposes.

Background Information

During the 10RFO which began on April 8, 1996, and ended on June 2, 1996, the hot leg tube-to-tubesheet expansion transition (i.e. roll transition) in tube 58-119 of Steam Generator A was examined. A single axial indication was detected in the roll transition as a result of the examination. The indication originated from the inside diameter and was purely axial in nature. There was no evidence of any circumferential component to the indication. The probe used to detect this axial indication was a 3-coil delta head with 0.115" diameter mid-range unshielded pancake coil, axial-directed coil, and circumferential coil. The roll transition was removed from the steam generator for laboratory destructive examination. Prior to the removal, an extensive in situ examination of the roll transition was conducted. This examination included the use of additional Steam Generator eddy current probes. Toledo Edison discussed the probe types used and the results of the examination to the NRC in TE Letter Serial Number 2391, dated July 23, 1996.

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Laboratory Examination

After the tube portion with the roll transition was removed from the steam generator, it was sent to Framatome Technologies (FTI) for laboratory examination. The tube was received by FTI in late June of 1996. The purpose of the laboratory examinations was to determine if the eddy current indication was caused by primary water stress corrosion cracking (PWSCC), a base material defect, or another degradation mechanism. If the indication was caused by a degradation mechanism, the laboratory examinations would characterize the mechanism and then attempt to determine the cause of the degradation. Several possible root causes were postulated: 1) Atypical tube material properties, 2) Re-rolling to the same depth as the original roll, which would have produced a non-stressed relieved roll transition susceptible to PWSCC, and 3) Aging of the tube material to the point of susceptibility to in-service corrosion.

A 160 degree arc of the tube roll transition containing the eddy current indication was sectioned for destructive examination by FTI. The remaining portion of the roll transition was retained for root cause analysis. The examination confirmed the presence of an axial crack corresponding to the field eddy current indication. Additionally, four smaller axial cracks were identified in the roll transition. The angular position, overall length and depth are summarized in the table below.

Angular Position (degrees)	Overall Length (inches)	Depth (% TW)	Remarks
82	0.050	43	
95	0.092	78	Axial crack assumed to correlate with the field eddy current indication.
109	0.014	N/A	Not analyzed due to similarity to the crack located at 176 degrees.
176	0.014	7	
190	0.043	43	

NOTE: Cracks were defined as those flaws which were sufficiently large enough to be detectable after flattening and examining under a stereo-microscope at magnifications up to 50X.

None of these five cracks were visible until the tube section was flattened, which then caused the cracks to open. All of the cracks were contained within the roll transition, were axially oriented (no circumferential extent), and exhibited essentially no branching.

Cracking Mechanism

The cracking mechanism was defined as PWSCC for the following three reasons: 1) The cracking was observed to be intergranular (corrosion induced), 2) the cracking occurred in the roll transition where high tensile residual stress was suspected, and 3) there was a lack of any obvious corrosive species to have caused the cracking.

In addition to the cracking, a 0.060" wide circumferential band of shallow intergranular attack (IGA) was visible throughout the section of the roll transition destructively examined. The IGA was typically less than 3% through-wall. There were numerous intergranular penetrations within the roll transition which were slightly deeper than the IGA. The penetrations were up to 8% through-wall. (Note that shallow IGA is typically observed on the inside diameter surface of once through steam generator (OTSG) tubing and has been attributed to the pickling process that the tubing undergoes during fabrication.)

Root Cause Analysis

The analysis of the tube found the tube's material properties were consistent with typical OTSG tubing. The tube material was tested for element composition, sensitization level, tensile strength, grain size and carbide distribution.

The analysis to determine if the tube was re-rolled after stress relief to the same depth as the original roll, required the use of mockup specimens. Mockup specimens of typical rolled tubes and tubes re-rolled to the same depth as the original roll following stress relief (i.e. re-rolled tubes) were made. Micro-hardness, residual stress, and cold work data from tube 58-119 and the mockup specimens was compared to determine if the data correlated better with the typical rolled tubes with post-roll stress relief, or with re-rolled tubes without post-roll stress relief. There was not a precise match of the data for tube 58-119 to either of the mockup specimens. However, the data for tube 58-119 did have a high correlation with the re-rolled tube data without post-roll stress relief obtained from mock-up testing. This high correlation is considered a sufficient demonstration that tube 58-119 was re-rolled to the same depth following stress relief during initial construction and did not receive a final post-roll stress relief. Also, the need for higher than anticipated pull loads to remove tube 58-119 from the steam generator upper tube sheet supports the conclusion that the tube was re-rolled.

There were 8 tubes originally believed to be non-stress relieved. A later re-review of records by FTI during the 10RFO indicated the roll transitions were stress relieved. This condition was previously described by TE in Letter Serial Number 2378, dated June 3, 1996 to the NRC. The data acquired by the recent destructive examination of the segment of tube 58-119 now indicates that tube 58-119 was re-rolled to some extent during initial construction without post-roll stress relief, and, new residual stresses were reintroduced by the re-roll process. Therefore, Toledo Edison believes that this roll transition is non-stress relieved.

The stress state of the other seven tubes examined during 10RFO has not been definitely established, however Toledo Edison is conservatively considering them to be re-rolled to some extent and will, therefore, examine them during future scheduled steam generator eddy current inspections.

<u>Regulatory Guide 1.121</u>

A Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," analysis was performed by FTI. The analysis concluded that Regulatory Guide limits were not exceeded. Tube burst is precluded by the restraint of the tubesheet, therefore, the tube burst pressure is larger than three times the normal operating pressure.

Three analyses were performed for primary-to-secondary leakage. The first analysis assumed the five cracks were each 100% through-wall with a tube wall thickness of 0.034" (minimum wall thickness is 0.034"). The leak rate was calculated to be less than 0.0005 gpm. The second analysis assumed the five cracks were each 100% through-wall and the tube wall thickness was reduced to 0.031" due to a band of IGA. This leak rate was calculated to be less than 0.001 gpm. The third analysis was very conservative to determine the maximum leak rate for a full 360 degree severance of a tube at the hot leg roll transition with the tube in tension. (Note that the tubes are in compression during normal operation and would be in tension during a main steam line break). This leak rate was calculated to be approximately 6 gpm. This leak rate is applicable for both normal operation and accident conditions because the flow is choked. All of the above leak rates are within the steam generator tube rupture accident analysis value of 435 gpm. Also, the plant primary-to-secondary leakage limit of 1 gpm would not be exceeded during normal operation, even if all five of the cracks were 100% through-wall.

<u>Future Inspection Plans</u>

The next scheduled eddy current examinations are during Davis-Besse's Eleventh Refueling Outage presently scheduled for April - May, 1998. The inspection scope for this outage will be determined as industry information further evolves concerning steam generator tube inspection techniques and degradation mechanisms.

There are three FTI reports which provide the details of the above information. These three reports are currently under final review by TE and the B&W Owners Group. All three reports are anticipated to be finalized by May 1, 1997.

Should you have any questions or require additional information, please contact Mr. James L. Freels, Manager - Regulatory Affairs, at (419) 321-8466.

Very truly yours,

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FWK/laj

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