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U.S. Nuclear Regulatory Commission
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Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Response to NRC Generic Letter 95-03 "Circumferential Cracking of Steam Generator Tubes"

Recent steam generator tube inspections at the Maine Yankee Nuclear Plant identified a large number of circumferential indications at the top of the tubesheet region. These inspection results prompted the NRC to issue Generic Letter 95-03 "Circumferential Cracking of Steam Generator Tubes". The attachment to this letter provides Wisconsin Public Service Corporation's (WPSC) response to the information requested by the Generic Letter.

If you have any questions or require additional information please contact a member of my staff.

Sincerely,

M Marchi

C. R. Steinhardt for
Senior Vice President - Nuclear Power

SLB/jrk

Attach.

cc - US NRC Region III
US NRC Senior Resident Inspector

Subscribed and Sworn to
Before Me This 27th Day
of June 1995

Jeanne M. Ferris
Notary Public, State of Wisconsin

My Commission Expires:
June 13, 1999

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ATTACHMENT 1

Letter from C.R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

June 27, 1995

WPSC Response to Generic Letter 95-03

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1.0 INTRODUCTION

Recent nondestructive examination of the steam generator (SG) tubing at the Maine Yankee Nuclear Plant has identified a large number of circumferential indications at the top of the tubesheet region. These most recent inspection findings, coupled with previously documented inspection results regarding circumferential cracking, have led to the issuance of NRC Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes" dated April 28, 1995. The information detailed herein will address the requested actions of the Generic Letter 95-03 as they pertain to the Kewaunee Nuclear Power Plant (KNPP) SGs.

The most recent inspection findings concerning SG tube expansion regions (Maine Yankee and Arkansas Nuclear One Unit 2) appear to have impacted those SGs utilizing the Combustion-Engineering (C-E) EXPLANSION process more than others. While there are similarities between the C-E EXPLANSION process and the Westinghouse WEXTEX process, the degree to which the Westinghouse units, regardless of tube expansion process, have been affected by circumferential cracking is significantly less than the most recent experience of the C-E units. Furthermore, the reported sludge pile height at Maine Yankee (up to 18 inches) may have influenced indication detectability. Such sludge pile thicknesses are not representative of currently operating Westinghouse units.

Successive inspection results using Motorized Rotating Pancake Coil (MRPC) probes for Westinghouse plants with hardrolled or explosively expanded tubes have indicated steadily declining numbers of new indications, declining angular extent and very low growth rates. The only occurrences of significant levels of circumferential cracking have been found when plants perform their first large scale MRPC inspection.

Available historical information shows that for some Westinghouse plants circumferential cracking has been detected in the tubesheet region tube expansion transitions from expanded to unexpanded tube, at the Row 1 and 2 U-bend tangent points, and at one plant (two twin units), at dented tube support plate intersections. The main focus of this response will be to address tubesheet region expansion transition cracking since this was the primary reason for the issuance of the generic letter. Other circumferential crack initiation sites will be addressed in the following sections due to the limited number of field indications detected and limited number of tubes which can be affected (specifically small radius U-bends and dented tube support plate intersections).

Collectively, the items discussed above and further detailed in the successive pages combined with the use of qualified eddy current inspection techniques and properly implemented analysis criteria provide for the safe continued operation of KNPP.

2.0 OPERATING EXPERIENCE ASSESSMENT

The Kewaunee Nuclear Power Plant (KNPP) steam generators are Westinghouse model 51. The SGs are constructed of mill-annealed inconel 600, 7/8" OD X 0.050 inch nominal wall thickness tubing with a partial depth hardroll expansion into the tubesheet. The SGs have been in service since plant start-up in 1974.

The KNPP SGs have been experiencing tube degradation attributed to axially oriented, outside diameter intergranular attack and stress corrosion cracking (ODSCC) in both the tubesheet and tube to tube support plate crevice regions. The characterization and orientation of this degradation form has been confirmed by pulled tube data from 1990 and 1993. As a result of this degradation significant tube plugging and sleeving has been required. During the 1988 and 1989 outages a large scale preventative and corrective sleeving program was implemented in the hot leg tubesheet crevice region. All of the sleeves installed were Westinghouse mechanical sleeves with a hybrid expansion joint (HEJ). Additional sleeving occurred in 1991 using the Westinghouse HEJs, and in 1992 with C-E welded sleeves.

During the 1994 refueling outage a motorized rotating pancake coil probe (MRPC), referred to as the I-coil, was used to inspect 100% of the Westinghouse HEJ upper sleeve joints and a sample of the C-E welded sleeve upper joints. A total of 77 parent tube indications were detected in the Westinghouse upper sleeve joints and no indications were detected in the C-E sleeve joints. The majority of these indications were reported as circumferential cracks located at, or within the top of the lower hardroll transition. During the 1995 refueling outage 100% of the HEJ upper joints were inspected with the MRPC plus-point probe. A total of 749 parent tube indications were detected, the majority of these were reported as circumferential cracks located at or within the top of the lower hardroll transition.

During the 1995 outage, three HEJ upper joint sleeve samples were removed from the B SG for further evaluation of the indications. The nondestructive part of the evaluation has been completed and confirms that the flaws are a network of short, tight, semi-continuous circumferentially orientated cracks located in the mid- to upper portion of the lower hardroll transition. The destructive evaluation of these sleeve joint samples is in progress and should be completed later this summer.

The operating experience of steam generators similar in design to those installed at KNPP as well as other designs has been reviewed in regards to the incidence of circumferential cracking. Based on this review, the regions of the KNPP tube bundles that are potentially susceptible to circumferential cracking are the tube to tubesheet expansion, the row 1 and 2 U-bends and sleeve joints.

Provided below are two tables of plant groupings. Table 1 lists the other Westinghouse plants using a similar SG tube expansion process and Table 2 lists those plants that have Westinghouse sleeves in service. Following these tables is a safety assessment justifying continued operation based on our evaluation of recent operating experience, a summary of future inspection plans and a schedule for the next SG inspection. This information is provided in response to NRC requested information items (1) and (2).

TABLE 1

Partial Depth Hardroll Expansion Plants
Alloy 600 Mill Annealed (MA) Tubing

Plant/Steam Generator Model	Startup	First Time Circ. Cracking	Location	Tube Pull and Results
Connecticut Yankee/27	1968	None	N/A	N/A
Cook Unit 1/51	1975	8/92	Top of Tubesheet Dent	Yes, Axially dominated cellular bands of ODSCC
Ginna/44	1970	Unknown	Roll Transition	Yes, unknown
Indian Point Unit 2/44	1973	3/95	Roll Transition	(a)
Kewaunee/51	1974	None	N/A	Yes, no roll transition circ. SCC, axial SCC in crevice
Point Beach Unit 2/44	1972	None	N/A	N/A
Praire Island Unit 1/51	1974	None	N/A	N/A
Praire Island Unit 2/51	1976	None	N/A	N/A
Zion Unit 1/51	1973	None	N/A	N/A
Zion Unit 2/51	1974	None	N/A	N/A

(a): Westinghouse believes that these indications are bands of closely spaced axial cracks.

TABLE 2				
HEJ and Laser Welded Sleeved Tubes				
Plant/Steam Generator	Sleeve Inst.	First Time Circ. Cracking	Location	Tube Pull and Results
Kewaunee (HEJ)/51	1988, 1989, 1991	4/94	Parent Tube Upper HEJ Lower Roll Transition[(a)]	Yes, no destructive exam. results to date
Point Beach Unit 2 (HEJ)/44	1983, 1984	9/94	Parent Tube HEJ Lower Roll Transition[(a)]	No
Cook Unit 1 (HEJ)/51	1992	None (no RPC)	N/A	N/A
Zion Unit 2 (HEJ)/51	1988	None (no RPC)	N/A	N/A
Farley Unit 1 (LWS)/51	1992	None	N/A	N/A
Farley Unit 2 (LWS)/51	1992	None	N/A	N/A

(a): Nearly all indications detected at HEJ hardroll lower transition. A few indications were detected at the upper and lower hydraulic transitions.

3.0 SAFETY ASSESSMENT

3.1 Partial Depth Hardroll Expansion Plants

The incidence of circumferential cracking at roll transitions in partial depth hardroll plants has been negligible throughout the nuclear industry. In 1990, two tubes were removed from KNPP for examination of significant indications within the tubesheet crevice region. Destructive examination of tube samples detected axially oriented ODSCC in the non-expanded tube length within the tubesheet crevice region. No PWSCC or circumferential indication were found in the roll transitions. In 1992, a number of indications were reported at D.C. Cook Unit 1. Tube pull results from this unit revealed that the degradation morphology was OD initiated cellular corrosion dominated by axial cracking. The degradation was located at the top of the tubesheet and attributed to localized denting. In March of 1995, circumferential cracking in the transition region was reported at Indian Point 2. Westinghouse believes that these indications are bands of closely spaced axial cracks as opposed to circumferentially oriented degradation.

During the KNPP 1991 refueling outage, a 100% MRPC inspection was performed in the open (nonsleeved) tubesheet hot leg crevices. No circumferential indications were detected during this inspection. During the 1992, 1993 and 1994 outages, a 100% bobbin coil inspection was performed with confirmatory MRPC exams, and in 1995 another 100% MRPC inspection was performed. No circumferential indications have been detected as a result of these inspections. Based on the results of prior inspections, the confirmation of the degradation by pulled tube data from KNPP, and lack of circumferential oriented indications reported throughout the industry in partial depth hardroll expansion plants, no indications would be expected that could challenge tube integrity at the end of this current operating cycle.

3.2 U-Bend Cracking

The incidence of circumferential indications at the Row 1 and 2 U-bend tangent points has not been significant, both in numbers of indications and in reported MRPC angles. All active Row 1 and 30% of the Row 2 tubes at KNPP have been inspected with MRPC since 1989. While a small number of U-bend indications have been detected, none of the indications have been circumferentially oriented. All tubes with indications have been removed from service. Based on the low incidence of U-bend circumferential cracking in the industry and the results of prior inspections at KNPP, no indications would be expected that could challenge tube integrity at the end of this current operating cycle.

3.3 High Cycle Fatigue

The concerns for high cycle fatigue and sudden tube failure are documented in detail in NRC Bulletin 88-02. KNPP has been analyzed for this condition using NRC accepted methodology and has in place administrative controls to ensure continued compliance with commitments made for implementing the recommendations of Bulletin 88-02. Therefore, this potential mechanism of initiating circumferential cracking has been adequately mitigated and will not impact steam generator structural integrity.

3.4 Dented Tube Support Plate (TSP) Intersections

Dented TSP intersections are not prevalent at KNPP. Boric acid addition has been used since 1988 primarily for mitigation of TSP ODSCC, however a side benefit has been control of denting.

With the implementation of the 2 volt alternate repair criteria, all dents greater that 5 volts were inspected with MRPC during the 1995 refueling outage. Although the population of dents this size is very small, those examined with MRPC did not reveal any evidence of circumferential cracking.

The potential for circumferential cracking associated with dented tube support plate intersections is considered remote considering the history of secondary water chemistry control and the use of boric acid addition. Therefore, this degradation mechanism is not postulated to have any impact on steam generator structural integrity.

3.5 Sleeved Joints

Sleeve joint inspections have been performed at KNPP using the MRPC I-coil in 1994, and the MRPC plus-point in 1995. As discussed in the previous background section, a significant number of parent tube indications were reported during these inspections and three HEJ samples were removed for further flaw evaluation. All of the indications detected in the upper HEJ lower hardroll transition were removed from service by plugging during the 1994 and 1995 refueling outages.

Evaluations performed by Westinghouse of the upper HEJ indicates that tube mean stresses above the hardroll region are compressive, while below the hardroll region tube mean stresses are expected to be tensile in nature, thus supporting the location of the detected indications. Work was started by Westinghouse to support an alternate repair criteria for the parent tube flaws located in the lower hardroll transition. The proposed alternate repair criteria is supported by leakage test data of the hardroll joint, tube bundle integrity and the interference fit of the hardroll joint. The results of this analysis demonstrates that even if the parent tube completely severed, hop-off of the upper portion of the tube would be precluded by the tube bundle and support plates, and resultant leakage from such tubes during a postulated steam line break condition would be well within acceptable limits due to the interference fit of the hardroll joint.

In addition, the preliminary results of the HEJ sample testing confirms that indications are a network of short, tight, semi-continuous circumferential cracks located at the top, or within the lower hardroll transition with a high degree of structural integrity. The results of the tensile separation of the joint for specimens R2C32 and R2C54 reveal that the cracking is ID initiated, 40% to 90% through wall and 360° around the joint with ligaments. The pull forces required to separate the joints were over 10,000 pounds. Therefore, circumferential cracking at this location does not present a concern in regards to tube bundle structural integrity.

4.0 FUTURE INSPECTION PLANS

The next scheduled SG tube examination is in the fall of 1996. WPSC plans to continue with implementation of the most recent revision of the EPRI PWR Steam Generator Tube Examination Guidelines. Initial sample sizes established and justified by this document and other commitments will form the basis for inspection scope. Qualified supplemental inspection techniques will continue to be used to interrogate regions of the tube bundle where bobbin coil eddy current techniques have limited capability to assess tube condition and/or resolve anomalous results. Sample expansion will be in accordance with the more conservative of the above mentioned guidelines or plant Technical Specifications. Data analysts qualified to Appendix G of the EPRI PWR Steam Generator Tube Examination Guidelines will continue to be required for analysis along with the administration of site specific testing.

5.0 SUMMARY

WPSC has reviewed the applicable industry operating experience regarding circumferential cracking of SG tubes and has assessed the potential impact of such degradation on KNPP.

The past inspection scope and results of each susceptible region of the tube bundle have been reviewed in regards to susceptibility to circumferential cracking, threshold of detection, expected or inferred crack growth rates and other relevant factors. Those susceptible regions of the SG tube bundle have been inspected using sampling plans and qualified supplemental techniques in accordance with EPRI PWR Steam Generator Tube Examination Guidelines. The capabilities of these supplemental inspection techniques to adequately assess steam generator tube condition and thus ensure structural integrity have been demonstrated and recently reaffirmed to be qualified for the detection of ID (PWSCC) and OD (ODSCC) stress corrosion cracking irrespective of orientation-circumferential or axial.

The past KNPP SG tube inspection results provide evidence that circumferential cracking of the susceptible regions of the tube bundle does not present a concern in regards to tube structural integrity. Therefore, KNPP is safe to operate until its next scheduled examination.