Wisconsin Electric
POWER COMPANY

231 W Michigan, P.O. Box 2046, Milwaukee, WI 53201-2046 (414) 221-2345

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Document Control Desk
U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555

Ladies/Gentlemen:

DOCKETS 50-266 AND 50-301
RESPONSE TO GENERIC LETTER 95-03
CIRCUMFERENTIAL CRACKING OF STEAM GENERATOR TUBES
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

On April 28, 1995, the Nuclear Regulatory Commission issued Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes." The Generic Letter was issued to all holders of operating licenses or construction permits for pressurized water reactors (PWRs) to notify addressees of the recent steam generator tube inspection findings at Maine Yankee Atomic Power Station and the safety significance of these findings. The letter also alerts addressees of the importance of performing comprehensive examinations of steam generator tubes using techniques and equipment capable of reliably detecting degradation to which the steam generator tubes may be susceptible.

Licensees were requested to respond to Generic Letter 95-03 within 60 days in accordance with the requirements of 10 CFR 50.54(f) by providing the following information:

1. A safety assessment justifying continued operation until the next scheduled steam generator tube inspection based on operating experience with respect to the detection and sizing of circumferential indications.

2. A summary of the next steam generator tube inspection plans developed as they pertain to the detection of circumferential cracking.



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Included as an attachment to this letter is our response to the Generic Letter as it applies to Point Beach Nuclear Plant. We believe this letter satisfies the 60-day response requirement of Generic Letter 95-03. If you have any questions, please contact us.

Sincerely,

G M Krieser
for
Bob Link
Vice President
Nuclear Power

Attachment

cc: NRC Regional Administrator
NRC Resident Inspector

Subscribed and sworn before me
this 26th day of June, 1995.

Jacqueline Gouch
Notary Public, State of Wisconsin

My commission expires 10-27-96.

Attachment

RESPONSE TO GENERIC LETTER 95-03
CIRCUMFERENTIAL CRACKING OF STEAM GENERATOR TUBES
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

## BACKGROUND

### Point Beach Nuclear Plant, Unit 1

The Point Beach Nuclear Plant (PBNP) Unit 1 steam generators are Westinghouse Model 44F with thermally-treated, 7/8 inch outside diameter, Inconel 600 tubing. The tubes are hydraulically expanded to the full depth of the tubesheet. The Unit 1 steam generators have been in service since the completion of steam generator replacement in Spring 1984.

All volatile secondary chemistry has been utilized throughout all of the Unit 1 steam generators' service lives. Since the Unit 1 steam generators were replaced, tube corrosion has been minimal.

Past inspections have consisted of the normal industry-accepted scope. Results have been very favorable with only three tubes requiring repair due to tube wall degradation. The current Unit 1 steam generators have never been the cause of a forced outage due to tube leakage.

### Point Beach Nuclear Plant, Unit 2

The PBNP Unit 2 steam generators are Westinghouse Model 44 with mill-annealed, 7/8 inch outside diameter, Inconel 600 tubing. The tubes are partial-depth hardroll expanded in the tubesheet. The Unit 2 steam generators have been in service since plant start-up in 1972. Coordinated phosphate secondary chemistry had been utilized for approximately the first two years of the Unit 2 steam generators' service lives. The partially expanded tube-to-tubesheet interface in combination with the initial coordinated phosphate chemistry regime resulted in excessive axially-oriented, outside diameter intergrannular attack and stress corrosion cracking (ODSCC) in the tubesheet region which necessitated the installation of Westinghouse mechanical hybrid expansion joint (HEJ) sleeves.

In November 1994, PBNP inspected the sleeved regions of the Unit 2 sleeved steam generator tubes using the new Westinghouse CECCO-5 probe. The CECCO-5 probe was developed specifically for detecting circumferential cracks in steam generator tubes and was used for the first time at PBNP. Of PBNP Unit 2's 3001 in-service sleeved tubes, 222 contained circumferential indications and were removed from service.

INDUSTRY OPERATING EXPERIENCE

Provided below are three tables of plant groupings. Tables 1 and 2 list other Westinghouse plants using similar tube expansion processes as our Unit 1 and Unit 2 steam generators, respectively. Table 3 lists those plants that currently have Westinghouse sleeved tubes in service.

Table 1 - Plants With Hydraulically Expanded Steam Generators Utilizing Inconel 600 Thermally Treated (TT) Tubing

| Plant/Steam Generator Model | Startup | First Time Circ. Cracking | Location | Tube Pull and Results |
|---|---|---|---|---|
| Braidwood Unit 2/D5 | 1988 | None | N/A | N/A |
| Byron Unit 2/D5 | 1987 | None | N/A | N/A |
| Callaway [a]/F | 1984 | None | N/A | N/A |
| Catawba Unit 2/D5 | 1986 | None | N/A | N/A |
| Comanche Peak 2/D5 | 1993 | None | N/A | N/A |
| Millstone Unit 3/F | 1986 | None | N/A | N/A |
| Point Beach Unit 1[b]/44F | 1984 | None | N/A | N/A |
| Robinson Unit 2 [b]/44 | 1984 | None | N/A | N/A |
| Seabrook/F | 1989 | None | N/A | N/A |
| Surry Unit 1 [b]/51F | 1980 | none | N/A | Yes, no SCC found, indication attributed to probe liftoff, tube geometric effects |
| Surry Unit 2 [b]/51F | 1981 | None | N/A | N/A |
| Turkey Point Unit 3[b]/44F | 1982 | None | N/A | N/A |
| Turkey Point Unit 4[b]/44F | 1983 | None | N/A | N/A |
| Vogtle Unit 1/F | 1987 | None | N/A | N/A |
| Vogtle Unit 2/F | 1989 | None | N/A | N/A |
| Wolf Creek/F | 1985 | None | N/A | N/A |

(a): Rows 1 through 10 are Alloy 600 TT only.
(b): Replacement Steam Generators.

Table 2 - Plants With Partial-Depth Hardroll Expansion Steam Generators Utilizing Inconel 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[a] | Roll Transition | None |
| 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 |
| Prairie Island Unit 1/51 | 1974 | None | N/A | N/A |
| Prairie 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): It is believed that these indications were bands of closely spaced axial cracks as opposed to circumferentially oriented degradation.

Table 3 - Plants With In-Service Hybrid Expansion Joint (HEJ) and Laser-Welded (LWS) Sleeved Tubes

| Plant/Steam Generator Model | Sleeve Inst. | First Time Circ. Cracking | Location | Tube Pull and Results |
|---|---|---|---|---|
| Kewaunee (HEJ)/51 | 1987, 1988, 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 Upper HEJ Lower Roll Transition[a] | None |
| 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.

RESPONSE TO NRC INFORMATION REQUEST

Full-Depth Hydraulic Expansion - Inconel 600 TT

The past two inspection programs for PBNP Unit 1 have been performed consistent with the EPRI and industry guidelines. There have been no reported domestic instances of any sort, axial or circumferential, in the expansion transitions of thermally treated tubing. Postulated end-of-cycle (EOC) crack angles would be projected to be well below the EOC structural limits for single throughwall cracks with 50 percent degradation in the remaining ligament. Since there is no domestic operating experiences of circumferential indications in full-depth hydraulically expanded steam generators with Inconel 600 TT tubing, it appears that growth rates of postulated circumferential indications are negligible. When considering these negligible growth rates and when factoring in the industry accepted detection thresholds for throughwall

circumferential degradation, no indications would be expected at PBNP Unit 1 which would challenge tube integrity at the end of the current operating cycle. Similarly, tube structural integrity is expected to be maintained during all future operating cycles.

Partial-Depth Hardroll Expansion

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 Kewaunee Nuclear Power Plant (KNPP) for examination of significant indications within the tubesheet crevice region. Destructive examination of the tube samples detected axially-oriented outside diameter stress corrosion cracking (ODSCC) in the non-expanded tube length within the tubesheet crevice region. No primary water stress corrosion cracking (PWSCC) or circumferential indications 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 outer diameter-initiated cellular corrosion dominated by axial cracking. The degradation was located at the top of the tubesheet and attributed to localized denting. In March 1995, circumferential cracking in the transition region was reported at Indian Point Unit 2. Based on analysis, it is believed that these indications were bands of closely spaced axial cracks as opposed to circumferentially oriented degradation.

During the Fall 1994 PBNP Unit 2 refueling outage, a 100% inspection of the tubesheet hot leg crevice region of unsleeved tubes was performed. No circumferential indications were detected as a result of this inspection. Based on the results of this most recent inspection, the results of the 1990 KNPP pulled tube data, and lack of circumferential oriented indications reported throughout the industry in partial-depth hardroll expansion steam generators, no indications are expected that could challenge tube integrity at the end of this current operating cycle.

HEJ Sleeved Tubes

Sleeve inspections have been performed on the PBNP Unit 2 steam generators using the CECCO-5 probe in 1994, and on the KNPP steam generators using the I-coil in 1994 and the Plus-Point probe in 1995. A significant number of parent tube indications were reported during these inspections. In addition, three HEJ samples were removed from the KNPP steam generators in 1995 for further flaw evaluation. All of the indications detected in the upper HEJ were removed from service by plugging during the 1994 and 1995 refueling outages.

Evaluations performed by Westinghouse of the upper HEJ indicate that tube mean stresses above the hardroll region are compressive, while below the hardroll region tube mean stresses are tensile in

nature, thus supporting the location of the detected indications. Work was performed to support an alternate repair criteria for the parent tube flaws located in the lower hardroll transition. We believe 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 that 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 results of the non-destructive examination (NDE) portion of the removed KNPP specimen examinations confirm that the cracks are a network of short, tight, semi-continuous circumferential cracks located at the top, or within, the lower hardroll transition. Preliminary tensile load test results show joint integrity in these specimens to exceed the requirements of Regulatory Guide 1.121, "Bases For Plugging Degraded PWR Steam Generator Tubes."

Finally, as presented during our meeting with NRC staff on June 6, 1995, Wisconsin Electric anticipates replacing the PBNP Unit 2 steam generators during Fall 1996. The new Unit 2 steam generators should ensure the same successful performance as we have experienced with the Unit 1 replacement steam generators.

## FUTURE STEAM GENERATOR INSPECTION PLANS

### Point Beach Nuclear Plant, Unit 1

The next scheduled steam generator inspection for Unit 1 is Spring 1997. During this inspection, PBNP will follow the EPRI recommended practices for sample selection, NDE techniques, and data analysis.

### Point Beach Nuclear Plant, Unit 2

The Unit 2 inspection plans, as presented during our February 1, 1995, meeting with NRC staff, are to inspect 20% of the sleeved tubes using a very discriminating probe, such as the Plus-Point or CECCO-5. Full-length inspection of 100% of the non-sleeved tubes, inspection of the sleeved tubes in the unsleeved portion with a bobbin coil, and any supplementary MRPC as required by the results of the bobbin inspections will be performed based on recent examination results. If necessary, additional tube inspections will be conducted in accordance with the criteria in PBNP Technical Specifications. Any indications noted by use of the motorized probes mentioned above and indications beyond the 40% plugging limit will be considered defective and be cause for remedial action. The analysts will be qualified per the PBNP-specific

guidelines as outlined in Appendix G of EPRI Report NP-6201, "PWR Steam Generator Examination Guidelines," Revision 3, dated November 1992. All probes used in the detection of circumferential cracks will be the latest state-of-the-art probes available which are qualified in accordance with Appendix H of EPRI Report NP-6201, Revision 3.

CONCLUSION

Our efforts have kept pace with, or even preceded, the industry. Previously performed examinations on the roll transitions of PBNP non-sleeved tubes using state-of-the-art probes and analysis techniques have relegated the type of indications we have experienced to being axially oriented with an outside diameter origin. The twenty-plus years of inspection experience has not shown signs of any inner diameter origin defects. The lessons learned by the most recent developments in the industry and the cumulative experience of the most recent ten years of operation go into each inspection plan and are also a basis for monitoring operational parameters. Our ability to maintain essentially defect-free Unit 1 steam generators with negligible leakage indicates an effective PBNP inspection program. Point Beach has historically maintained conservative mitigating actions such as the reduction of reactor coolant system pressure, reduction of reactor coolant system average temperature, the implementation of advanced amine (ETA) chemistry control, and conservative abnormal/emergency operating procedures (AOPs and EOPs) which are designed to mitigate all steam generator tube rupture events regardless of the causal factors. Point Beach also performs an 800 psi differential pressure test during each inspection outage, monitors secondary side steam generator molar ratio, and performs continuous primary-to-secondary leak rate monitoring. The monitoring of these parameters has shown us to be effective in the inspection and maintenance activities and even more effective in the conservatism necessary to operate free of mid-cycle shutdowns related to steam generator tube degradation.