



Wisconsin Electric POWER COMPANY
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December 21, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Attention: Mr. A. Schwencer, Chief
Operating Reactors Branch 1

Gentlemen:

DOCKET NO. 50-266
STEAM GENERATORS
POINT BEACH NUCLEAR PLANT, UNIT 1

On November 5 and November 20, 1979, meetings were held in Bethesda with representatives of the NRC Staff to present information and data related to recent Unit 1 steam generator tube problems. This information was confirmed by our letter to the NRC of November 23, 1979.

Our Licensee Event Report (LER) No. 79-017/01T-C, dated November 16, 1979, provided information in regard to eddy current examination results obtained during the most recent Unit 1 refueling in October. A listing of plugged tubes, defect size and defect location for each steam generator was included in the LER. Among other things, this LER indicated there were two tubes in "A" steam generator and two tubes in "B" steam generator with defects at the top of the tubesheet and one tube in the "B" steam generator with a defect one-half inch above the tubesheet. This letter is in response to your Mr. Trammell's request for further information in respect to these five tubes.

LER No. 79-017/01T-0 provided details of the steam generator eddy current October 1979 inspection program, the results of the inspection and the location of plugged tubes and stated that three tubes had been physically removed for metallurgical analysis. The results of the metallurgical analyses on these three tubes were discussed with your Staff on November 20 and documented in our November 23 letter. Your attention is called to the observations and conclusions on page 3 of that letter where it is stated:

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- "4. Based on samples removed from the Unit 1 steam generator and from other plants in previous years, these defects are concluded to be intergranular corrosion attack of the Inconel 600 tube material in the narrow crevice area caused by residue of caustic materials remaining from phosphate chemical treatment and possibly from earlier condenser tube leakage.
- "5. This corrosion is confined to the crevice area. No evidence of this intergranular attack was found in areas of tube above the tubesheet."

Our conclusion in respect to the absence of IGA above the tubesheet is based on examinations of samples of tubes removed, as stated on pages E3-3 and E3-4 and Viewgraphs 5, 16 and 23 of Enclosure 3 to our November 23 letter. This conclusion is supported further by the burst tests summarized in item 7, page 4 and detailed on pages E3-3 through E3-6, where on page E3-6, the following statement was made:

- "2. The difference in burst strength between the specimens from deep in the crevice and that which spans the tubesheet indicates essentially no degradation in burst strength in the region of the tubesheet from a virgin tube burst strength. This coincides with the result of the metallurgical examination which concluded that there was no intergranular attack above this tubesheet."

During our meetings, we advised you of the limitations of available eddy current techniques to detect IGA. On page 5 of our November 23 letter, we stated the following:

- "9. Present eddy current testing has demonstrated its capability to detect cracks from a range of 20% through wall to full-wall penetration. This technique, however, is not an effective means of detecting intergranular corrosion in the absence of cracks. Because a diametrically unrestrained tube, such as in the space above the tubesheet and outside tube support plates, expands under internal pressure, intergranular corrosion in such areas would be revealed as cracks. In the tightly constrained areas within the tubesheet, tube expansion under internal pressure is effectively precluded. While this makes eddy current indication of intergranular attack incapable, the totally contained tube cannot expand under internal pressure to cause significant cracking or rupture during normal or accident conditions."

Confirmation of the eddy current method of crack detection was made by metallographic examination as detailed on page E3-3 and Viewgraph 5 of Enclosure 3 to our letter of November 23. However, no correlation could be found between eddy current test results and presence of IGA.

On this basis, we made no attempt to relate the eddy current inspection results contained in LER 79-017/01T-0 with the presence or absence of IGA. It is our opinion, and that of our consultants, that traditional eddy current inspection techniques make such a correlation unsupported. You may recall that Westinghouse Electric Corporation stated at our November 20 meeting that they would be seeking to make future improvements in eddy current testing methods in order to demonstrate ability to detect IGA by this technique.

The five tubes which had eddy current indication at or above the tubesheet during the October 1979 inspection are identified as follows:

<u>Steam Generator</u>	<u>Tube</u>		<u>Eddy Current Indication</u>
	<u>Row</u>	<u>Column</u>	
A	22	46	55% defect at top of tubesheet
A	30	57	80% defect at top of tubesheet
B	28	38	45% defect at top of tubesheet
B	30	44	83% defect at top of tubesheet
B	32	42	61% defect 1/2" above tubesheet

In October 1979, these five tubes, and all others, were inspected utilizing multi-frequency eddy current techniques. As a result of the defects indicated, these tubes were, of course, plugged, as reported in LER 79-017/01T-0. As we stated above, we made no attempt to relate these defects to IGA. We had included them in the "crevice corrosion" column of Viewgraph 1, in our November 23 letter based upon preliminary information regarding the number of tubes containing defects and, as we have since determined, this was in error.

These five tubes are in the same area of the tube bundle where many similar indications at or slightly above the tubesheet were observed in the early years of Unit 1 operation. These multiple indications were first reported to you in our letter dated April 7, 1975, in this docket. All of the defects were attributed to thinning or cracking. Since our November 20 meeting,

we have reviewed the single frequency eddy current test results since 1975 for these tubes and compared the signals of these past inspections to the same frequency signal obtained during the multi-frequency inspection in October. This comparison shows that the signals have not changed through three or four annual inspections since 1975.

The use of the multi-frequency technique in the October 1979 inspection, however, allowed discrimination of the eddy current tube defect signal from tubesheet interference such that identification of defects was made in these five tubes. Based on these reviews, we believe that the defects which have apparently remained essentially unchanged since at least 1975 resulted from earlier thinning or cracking rather than to the IGA currently being experienced in the tubesheet crevice area. The inclusion of these tubes in the "crevice corrosion" column of Viewgraph 1, attached to our November 23 letter, is now, in the light of this comparison, incorrect. In addition to this correction, we note that an outage in May 1978 to plug one tube in steam generator A was omitted from Viewgraph 1.

We enclose for your information a revised table which reflects these corrections. It should be noted that, while the two tubes in steam generator A and three tubes in steam generator B are indicated as being plugged due to cracking or thinning in October 1979, the defects apparently occurred in 1975 or earlier based upon review of the previous inspection results.

We believe that the accuracy within which eddy current indications can be axially located is approximately plus or minus one-half inch. This estimated accuracy is based on the size of the eddy current probe, the location of the two coils which are 1/8" apart on the probe, the precision of the axial measurement of the location of the probe in reference to the tube end and entrance and exit effects.

In response to a further question from Mr. Trammell, the metallographic examination indicated no IGA present within 3/4" of the top of the tubesheet as reported in pages E3-2, 3 and 4 and Viewgraphs 5, 10 and 16 of Enclosure 3 to our letter of November 23, 1979.

You are also aware that Unit 1 was shut down on December 11, following detection of steam generator tube leakage of about 250-260 gallons per day, in accordance with Section IV, item 3 of the Commission's Confirmatory Order of November 30. We are making a separate report in respect to this event, but the occasion of this shutdown also permitted us to make a more sensitive eddy current examination of the areas of concern. Although not required by our Technical Specifications or the Confirmatory Order of December 1, examination of about 1900 tubes in both A and B steam generators revealed no eddy current indications at or above the tubesheet.

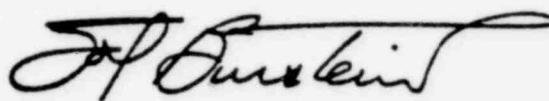
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In summary, we would reiterate that all physical examination and observations indicate intergranular attack is confined to the tubesheet crevice area. This is based on examination of tubes removed from steam generators. Present eddy current techniques are capable of detecting tube cracks but cannot be relied upon to indicate IGA. Any eddy current indication at or near the top of the tubesheet can, if sufficiently unscrambled from tubesheet interference, indicate a crack, but does not indicate IGA. We believe our conclusions, and those in the NRC Safety Evaluation Report dated November 30, are not affected by these unrelated concerns regarding eddy current indications in these five tubes.

Please advise us if you have any further questions.

Very truly yours,



Executive Vice President

Sol Burstein

Enclosure

Blind Copy to Messrs. C. S. McNeer
R. H. Gorske/A. W. Finke
C. W. Fay
D. K. Porter
G. A. Reed
Gerald Charnoff

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UNIT 1 STEAM GENERATOR
TUBE PLUGGING HISTORY

Date of Outage	Elapsed Time (Years)	Tubes Plugged										Cumulative Percent	
		Denting		Thinning or Cracking		Crevice Corrosion		Other		Total		A	B
		A	B	A	B	A	B	A	B	A	B		
12/21/70	0	-	-	-	-	-	-	1(1)	-	1	-	<0.1	0
9/30/72	1.8	-	-	87	91	-	-	14	4(2)	102	95	3.1	2.9
4/6/74	3.3	-	-	1	1	-	-	-	-	103	96	3.2	2.9
2/26/75	4.2	-	-	59	98	-	-	-	-	162	194	5.0	6.0
11/16/75	4.9	-	-	6	4	-	-	-	-	168	198	5.2	6.1
10/1/76	5.8	-	-	-	-	-	-	-	-	168	198	5.2	6.1
6/24/77	6.5	-	-	-	1	-	-	-	-	168	199	5.2	6.1
10/4/77	6.9	10	-	-	-	1	2	-	-	179	201	5.5	6.2
2/1/78	7.1	-	-	-	-	-	-	1(3)	-	180	201	5.5	6.2
5/26/78	7.4	-	-	1	-	-	-	-	-	181	201	5.5	6.2
9/20/78	7.7	1	-	-	-	6	4	-	-	188	205	5.7	6.3
3/1/79	8.2	-	-	-	-	8	1	-	-	196	206	6.0	6.3
8/5/79	8.6	-	-	-	-	52	45	-	-	248	251	7.6	7.7
8/29/79	8.8	-	-	-	-	2	-	2(4)	-	252	251	7.7	7.7
10/5/79	8.9	-	-	2	3(6)	68	61	7	4(5)	329	319	10.1	9.8

Notes: (1) Plugged during manufacture.

(2) Fourteen tubes in A were plugged due to gouging during machining for clad repair. Three tubes in B were removed for analysis and one was plugged by mistake.

(3) Plugged tube was in periphery.

(4) An audit of tubesheet photographs indicated two tubes which were plugged but previously not included in inspection reports.

(5) Seven tubes in A included three with defects less than the plugging limit, two tubes which had no indications but which were pulled for analysis, and two tubes plugged by mistake. Four tubes in B included three tubes with indications less than the plugging limit and one tube plugged by mistake.

(6) Two tubes in A and three tubes in B were plugged due to defects identified at or above the tubesheet using multi-frequency eddy current techniques. These defects are attributed to thinning or cracking in prior years, based upon comparison with single frequency eddy current results from previous inspections.

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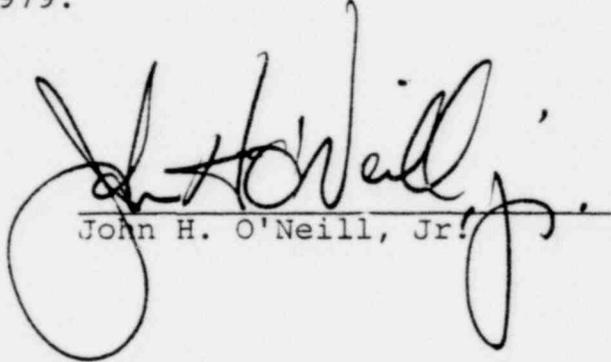
UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

In the Matter of)
)
WISCONSIN ELECTRIC POWER COMPANY) Docket No. 50-266
)
(Point Beach Nuclear Plant,) (Modification of License)
Unit 1))

CERTIFICATE OF SERVICE

I hereby certify that copies of "Licensee's Response to Request by Wisconsin's Environmental Decade, Inc. For Hearing on Confirmatory Order," dated December 27, 1979, were served upon those persons on the attached service list, by deposit in the United States mail, postage prepaid, or by hand delivery, on this 27th day of December, 1979.


John H. O'Neill, Jr.

Dated: December 27, 1979

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

the Matter of)
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 WISCONSIN ELECTRIC POWER COMPANY) Docket No. 50-266
)
(Point Beach Nuclear Plant,) (Modification of
Unit 1)) License)

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