NORTHEAST UTILITIES
THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

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July 3, 1990

Docket No. 50-336
B13556
Re: S/G ECT Inspection

Mr. Thomas T. Martin
Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Dear Mr. Martin:

Millstone Nuclear Power Station, Unit No. 2
Steam Generator Inspection

Introduction

In a letter dated February 7, 1990,(1) Northeast Nuclear Energy Company (NNECO) submitted an assessment of the Millstone Unit No. 2 steam generators (S/G) to the NRC Staff. The information in that submittal was further explained and supplemented by a meeting that was held on February 22, 1990, in the NRC Headquarters in White Flint, Maryland. On May 9, 1990,(2) NNECO submitted a letter reiterating the position that continued operation was fully justified and prudent.

On May 11, 1990, representatives of the NRC and NNECO met in the NRC Headquarters in White Flint, Maryland to discuss plans for a steam generator tube inspection at Millstone Unit No. 2 during the then-current shutdown. NNECO had originally been scheduled to discuss with the NRC Staff the basis for operation to the end of the current cycle (September 1990) without a steam generator tube inspection. However, due to an increasing trend of primary-to-secondary leakage from May 8 to 10, NNECO decided to go to cold shutdown and perform a steam generator tube inspection. The plant was at hot shutdown due to a manual shutdown to repair an automatic feedwater valve.

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(1) E. J. Mroczka letter to W. T. Russell, "Millstone Nuclear Power Station, Unit No. 2, Steam Generator Inspection," dated February 7, 1990.

(2) E. J. Mroczka letter to W. T. Russell, "Millstone Nuclear Power Station, Unit No. 2, Steam Generator Safety Assessment," dated May 9, 1990.

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Since that time, NNECO has conducted a second in-cycle inspection of S/G tubes susceptible to cracking. The purpose of this inspection was to identify any stress corrosion cracks which may have developed, to ascertain whether the corrosion process was proceeding as predicted, and to locate and repair the cause of primary-to-secondary leakage observed during operation. The conclusions presented in the February 7, 1990 letter were based on assessments of how the cracks form and grow, predictions of the number and size of cracks expected to be present by the end of the cycle, and the structural implications of these cracks. The following discussion addresses the results of the second in-cycle inspection and their implications with regard to safety and reliability of S/Gs for the remainder of the current operationg cycle.

Discussion

The second in-cycle inspection of S/G tubes consisted of a rotating pancake coil (RPC) inspection of 100 percent of the region considered susceptible to cracking and a 3 percent random sample of tubes outside the susceptible region. Tubes which were identified as potentially cracked were inspected with a three-coil RPC probe and by ultrasonic testing (UT).

A total of 23 tubes were identified as cracked. A structural integrity assessment of the 23 cracked S/G tubes was performed. The results of this assessment indicate that all of these tubes were acceptable per Regulatory Guide 1.121.

Attachment 1 provides the circumferential extent as indicated by RPC and the average depth as indicated by UT. Two methods were used to determine the average depth. The first method listed represents a newly developed refinement in the method of evaluating the obtained UT data and reflects the degraded area evaluated in 5-degree increments. The second method is based on assuming the maximum depth over the entire crack segment and is less accurate, tending to overcall the average depths of the larger volume cracks. This method was included for comparison purposes since the October 1989 inspection depths were determined by this method.

The distributions of circumferential extent and average depth (when determined by the same method as the October 1989 average depths) observed in the current interim inspection are virtually identical to the predicted gamma distributions. The number of cracks identified during the current interim inspection are bounded by the 45 cracks assumed to be present by EOC 10 in the Reference 1 assessment. The statistical inferences with regard to the potential for structurally significant cracks, which formed the basis of the previous conclusion that continued operation of the S/Gs was safe for the remainder of the cycle, remain valid. The agreement between the predicted and actual results indicate that the improvements in chemistry and boric acid treatment have been effective in controlling the corrosion.

During the recent shutdown, the source of the primary-to-secondary leakage observed during operation was investigated. Secondary side pressure tests

were performed and a total of six leak paths identified. All of the leak paths were in tubes which had been previously plugged (five) or sleeved (one). Therefore they were not associated with a circumferential crack in an active tube. The identified leak paths were repaired. The occurrence of minor leakage in previously repaired tubes is not unexpected and does not affect the safe operation of the steam generators.

The circumferential cracks identified in the Millstone Unit No. 2 steam generators are expected to exhibit a leak-before-break behavior, and the 0.1 gpm leakage limit previously established remains valid.

Conclusions

Continued operation of the Millstone Unit No. 2 S/Gs for the remainder of Cycle 10 has been evaluated and found to be safe in accordance with applicable regulatory guidelines.

Please contact us if you have any questions or need additional information.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

E. J. Mroczka
Senior Vice President

Attachment
cc: G. S. Vissing, NRC Project Manager, Millstone Unit No. 2
W. J. Raymond, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3
U.S. Nuclear Regulatory Commission, Document Control Desk

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Attachment 1

Millstone Nuclear Power Station, Unit No. 2

Summary of Crack Measurements
May 1990 Steam Generator Examination

July 1990

Millstone Nuclear Power Station, Unit No. 2
Summary of Crack Measurements
May 1990 Steam Generator Examination

SG Number	Side	Line	Row	RPC Crack Lengths, Degrees	Maximum UT Crack Depth, Percent	Average UT Crack Depth, Percent (Increment Method)	Average UT Crack Depth, Percent (Maximum Depth Method)
1	C	27	15	69	69	15	19
1	C	40	52	83	47	6	9
1	C	51	9	176	48	18	11
1	C	52	10	194	73	13	22
1	C	53	23	64	51	27	35
1	C	54	22	153, 56	66	28	41
1	C	70	34	76	46	7	8
1	C	112	10	61, 53	90	29	54
1	C	140	38	90	46	23	26
1	H	35	55	57	33	6	4
1	H	48	78	48	44	7	5
1	H	70	58	160	68	27	36
1	H	77	99	50	--(a)	--	--
1	H	97	99	154	--(a)	--	--
1	H	110	12	45	45	13	8
1	H	114	26	72	55	12	13
1	H	123	13	74	--(a)	--	--
1	H	126	26	40	--(a)	--	--
1	H	132	48	54	--(a)	--	--
1	H	137	21	32	45	18	7
1	H	140	32	88	74	22	29
1	H	141	29	40	76	31	36
2	H	71	83	36	NO TEST(b)	--	--

(a) UT identified flaw mechanism as pitting.

(b) UT probe became restricted before reaching area of interest.