CATEGORY 1

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SUBJECT: Forwards results of SG eddy current tube insp & subsequent SG tube repairs performed during 1997 Unit 1 refueling outage.

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May 3, 1997 AEP:NRC:1166AH

Docket No.: 50-315

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Gentlemen:

Donald C. Cook Nuclear Plant Unit 1
STEAM GENERATOR 15 DAY INSPECTION REPAIR REPORT

Pursuant to the requirements of Cook Nuclear Plant technical specification (T/S) 4.4.5.5a, this letter is to inform you of the results of our steam generator eddy current tube inspection and the subsequent steam generator tube repairs performed during the 1997 unit 1 refueling outage.

Eddy current tube inspection of the unit 1 steam generators began on March 16, 1997. All inspections and repairs were completed on April 9, 1997. Inspections included examination of all previously non-plugged tubes. Attachment 1 describes the scope of the examination and repair activities. Attachment 2 discusses the investigation of the defects found and the actions taken as a result of those investigations, in accordance with the follow-up requirements of T/S 4.4.5.5c for inspections that fall into category C-3.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

vlb

Attachments

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ATTACHMENT 1 TO AEP:NRC:1166AH

UNIT 1 STEAM GENERATORS
TUBE INSPECTION REPORT
INSPECTION RESULTS (C-3) AND TUBE REPAIRS

Table 1 provides a summary of the 1997 unit 1 refueling outage (U1R97) steam generator (SG) inspection findings. The results are presented for each SG, by degradation location, based upon technical specification (T/S) plugging criteria.

The primary contributing degradation mechanism for the defective tubes was outside diameter stress corrosion cracking (ODSCC) in the tubesheet crevice region at or near the top of the tubesheet, and primary water stress corrosion cracking (PWSCC) at the partial depth roll transition. Both the ODSCC and PWSCC degradation was axial in nature.

Tube repair methods employed during U1R97 included both hot leg tube end re-roll and plugging. A total of 699 tubes were kept in service via re-roll repair. A total of 979 tubes were plugged. This total included forty-five sleeved tubes and one tube that was plugged following a tube pull. Details of the repair activities are summarized in table 2.

T/S 4.4.5.5d for implementation of the interim plugging criteria for tube support plate intersections requires NRC notification prior to returning to service should any of the conditions noted in T/S 4.4.5.5d 1-4 arise. None of these conditions arose. In summary: 1) actual end-of-cycle voltage distribution did not result in exceeding the leak limit during the previous cycle; 2) no crack-like circumferential indications were detected at the support plate intersections; 3) no indication extended beyond the confines of the tube support plate; and 4) conditional burst probability does not exceed 1x10 E-2.

Table 1

Area	SG 11				SG 12				SG 13				SG 14			
	Ind	Defects	Recy Lst	Plg Lst	Ind	Defects	Recy Lst	Plg Lst	Ind	Defects	Recy Lst	Plg Lst	Ind	Defects	Recy Lst	Plg Lst
CLTSP	100	13	--	13	88	18	--	18	61	10	--	10	73	20	--	20
AVB	7	0	--	0	29	2	--	2	59	2	--	2	22	0	--	0
U-Bend	6	6	--	6	10	8	--	8	9	8	--	8	0	0	--	287
HLTS	272	243	0	243	100	89	0	89	235	209	0	209	300	287	0	0
HLTSP	370	0	0	0	171	0	0	0	160	0	0	0	359	0	0	0
HLRT	384	157	152	5	240	123	123	0	714	226	219	7	169	72	72	0
HLSLV	39	16	--	16	20	10	--	10	36	15	--	15	13	0	--	0
Misc	0	7	--	2	0	3	--	3	1	10	--	3	3	3	--	3
TOTAL	1178	442	152	285	658	253	123	130	1275	480	219	254	939	382	72	310

Recy Lst = recovery list
Plg Lst = plugging list
CLTSP = cold leg tube support plate intersection
AVB = anti-vibration bar intersection
U-Bend = tight radius (rows 1 & 2) u-bends

HLTS = hot leg tubesheet region
HLTSP = hot leg tube support plate intersection
HLRT = hot leg tubesheet roll transition
HLSLV = hot leg hybrid expansion joint sleeve

Table 2

	Re-Rolled Tubes U1R97	Remaining Sleeved Tubes In-Service	Plugged Tubes U1R97	Plugged Tubes Total
SG 11	198*	808	285	676
SG 12	134*	170	130	436
SG 13	295*	443	254	447
SG 14	72	373	310	606
Totals	699	1794	979	2165

* Totals include rework of selected tubes previously re-rolled in 1995

ATTACHMENT 2 TO AEP:NRC:1166AH

UNIT 1 STEAM GENERATOR TUBE INSPECTION REPORT
CATEGORY C-3 DEGRADATION INVESTIGATION
AND
CORRECTIVE ACTIONS

The 1997 unit 1 refueling outage (U1R97) steam generator (SG) eddy current inspection resulted in a category C-3 classification, per technical specification (T/S) 4.4.5.5. The primary reason for this classification was the defects identified in the hot leg tubesheet region of each SG. An additional basis for the C-3 classification were indications found in the low row U-bend area.

The contributing degradation mechanism for the defective tubes in the tubesheet region was outside diameter stress corrosion cracking (ODSCC) in the tubesheet crevice at or near the top of the tubesheet, and primary water stress corrosion cracking (PWSCC) at the partial depth roll transition.

The cause for the increase in defective tubes in the tubesheet was investigated primarily for ODSCC at the top of tubesheet, because this degradation form is of most concern from the aspect of RG 1.121 tube integrity for leak and burst. An historical data review was conducted and it was determined that all the significant indications were present in the previous (1995) inspection. In some cases, these indications were traceable back to the 1994 inspection.

In 1995, the inspection of the tubes in this region was conducted using a 'Cecco-5' eddy current probe. During the 1997 'plus point' inspection, it became apparent that the previous inspection had not identified all the repairable indications in the region. Therefore, the larger than projected increase in tubesheet defects in 1997 can be attributed to undetected defects from previous cycles. Subsequent re-analysis of the Cecco-5 data, using today's analysis techniques and guidelines, was able to more readily identify the presence of the ODSCC indications.

A re-analysis of the Cecco-5 data from 1995 was not conducted for PWSCC due to the conclusive results of the ODSCC re-analysis. It was concluded that the same results would be seen for the PWSCC.

The initial U-bend sample motorized rotating pancake coil inspection program encompassed inspection of all inservice row 1 and row 2 tubes in SG 11. This program was expanded following discovery of a leaking row 1 tube during the secondary side hydro test that was subsequently confirmed by eddy current inspection. The SG 11 inspection scope was expanded to include row 3 tubes. The general inspection program was expanded into inservice row 1 and 2 tubes in the remaining SGs. Overall inspection findings were essentially split with 12 circumferential indications and 13 axial indications. All tubes with U-bend indications were plugged.

No indications were noted in the U-bend region during an inspection in the 1995 unit 1 refueling outage. This inspection examined the inservice row 1 and row 2 tubes in SG 14. Likewise, the U1R97 inspection on SG 14 did not identify any indications. Review of plant leakrate history and current inspection results suggest that the low row indications developed over a number of fuel cycles. Based upon plant experience, problems in the low row U-bends have not been prevalent, which, in turn, suggests that it is a slow growth rate phenomenon.

Insitu pressure testing was conducted on the bounding tubes in the U-bend area to quantify acceptable leakage and assure structural integrity. The insitu testing results confirmed tube integrity requirements per RG 1.121 and acceptable leakage.

Prior to returning to service, all tubes with non-recoverable (via re-roll) defects were plugged. Table 2 of attachment 1 contains a breakdown of the repairs conducted.

Both ODSCC and PWSCC remain a significant concern in the unit 1 SGs. While the U-bend area is not termed a significant concern due to the limited number of tubes affected and the slow growth rate, it will nevertheless be an area of increased focus during upcoming inspections. Current remedial actions consisting of reduced temperature and pressure operation and application of industry-approved chemistry guidelines will remain in effect. Additionally, implementation of aggressive inspection programs and repair techniques will be maintained to mitigate the impact of these and other degradation mechanisms found in Cook Nuclear Plant SGs.