10CFR50.73

SCE&G

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John L. Skolds
Vice President
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April 16, 1993
Refer to: RC-93-0094

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

Gentlemen.

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
LER 93-002

Attached is Licensee Event Report No. 93-002 for the Virgil C. Summer Nuclear Station. This report is submitted pursuant to the requirements of Technical Specification Section 4.4.5.5.c and 10CFR50.73(a)(2)(ii).

Should there be any questions, please call us at your convenience.

Very truly yours,

John L. Skolds

John L. Skolds

ARR/JLS/nkk
Attachment

c: O. W. Dixon (w/o attachment)
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Central File System

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NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Virgil C. Summer Nuclear Station

DOCKET NUMBER (2): 0 5 0 0 0 3 9 5

PAGE (3): 1 OF 0 4

TITLE (4): Steam Generator Tube Eddy Current Results - Category C-3 Applied

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 3	1 8	9 3	9 3	0 0 2	0 0	0 4	1 6	9 3		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 6

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	X 50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
J. R. Proper, Supervisor, OE, Reg. Issues & Plant Support	8 0 3	3 4 5 - 4 0 8 8

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
				N					

SUPPLEMENTAL REPORT EXPECTED (14)

X YES (If yes, complete EXPECTED SUBMISSION DATE) | NO

EXPECTED SUBMISSION DATE (15): MONTH 1 0 DAY 1 5 YEAR 9 3

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On March 18, 1993, at approximately 1037 hours, the preliminary results of the seventh inservice eddy current examination of the Virgil C. Summer Nuclear Station Steam Generators (SGs) were obtained. The planned examinations included 100% of the available (i.e., unplugged) tubes in each SG hot leg tubesheet, utilizing Motorized Rotating Pancake Coil Eddy Current Testing. A 100% Bobbin Coil inspection of the full length tubing (Tube End Cold to Tube End Hot) was also performed. The results indicated that since more than 1% of the inspected tubes in each SG were defective, a C-3 inspection category, per Technical Specification 4.4.5.2, applied. When the first SG (SG B) entered Category C-3, a prompt notification was made to the NRC Operations Center in accordance with the requirements of Technical Specification 4.4.5.5.c and 10CFR50.72(b)(2)(i).

As a result of the inspections performed during the seventh refueling outage, plugs were installed as follows: SG A-181, SG B-255 and SG C-212.

The Category C-3 tube degradation is localized in the tubesheet area and is the result of Primary Water Stress Corrosion Cracking (PWSCC). In addition, a number of tubes experienced Outer Diameter Stress Corrosion Cracking (ODSCC) in the tube support plate area. Tube plugging was completed on April 9, 1993. Three tubes were pulled to confirm the degradation mechanism in the tube support plate area. A supplemental report with tube pull results will be provided by October 15, 1993.

PLANT IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

EQUIPMENT IDENTIFICATION:

Westinghouse Model D-3 Steam Generators (SG)
Reactor Coolant System - AB - EIIS

IDENTIFICATION OF EVENT:

Steam generator tube eddy current examinations yielded inspection Category C-3 results in the hot leg tubesheet area, i.e., greater than 1% of the inspected tubes in each generator were defective with defects localized in the tubesheet area. In addition, a number of tubes experienced degradation in the tube support plate area.

EVENT DATE: March 18, 1993 at approximately 1037 hours.

REPORT DATE: April 16, 1993

This is the followup report to the 10CFR50.72 notification required by Section 4.4.5.5.c of the Virgil C. Summer Nuclear Station Technical Specifications. The initial notification was made as required by Technical Specification 4.4.5.5.c upon entry into Category C-3. It was previously agreed between the NRC Resident Inspector (L. Keller) and South Carolina Electric & Gas Company (SCE&G) that only one phone call was necessary upon the first steam generator entering Category C-3.

This report was initiated by Off-Normal Occurrence report 93-022.

PREVIOUS SIMILAR EVENTS:

Previous eddy current inspections have yielded similar results in the tubesheet area and have been reported to the NRC in LER 85-031 submitted on November 13, 1985, LER 87-006 submitted on May 7, 1987, LER 88-011 submitted on November 23, 1988, LER 90-005 submitted on May 4, 1990, and LER 91-008 submitted on November 1, 1991.

CONDITIONS PRIOR TO THE EVENT:

0% Power - Seventh Refueling Outage

DESCRIPTION OF EVENT:

The seventh inservice eddy current examination of the Virgil C. Summer Nuclear Station steam generator tubing was performed during March 1993. The examination was performed in accordance with the ASME "Boiler and Pressure Vessel Code" Section XI, 1977 edition, through Summer 1978 addenda, and as specified by Regulatory Guide 1.83, Revision 1, July 1975. The examination was performed on 100% of the available tubing (i.e., unplugged) in the hot leg tubesheet region, utilizing Motorized Rotating Pancake Coil (MRPC) Eddy Current Testing techniques. A 100% Bobbin Coil inspection of the full length tubing (Tube End Cold to Tube End Hot) was also performed on all available tubes.

Also, in anticipation of the need for an Interim Plugging Criteria (IPC) for the tube support plate area the inspection guidelines of WCAP-13522 "V. C. Summer Steam Generator Interim Tube Plugging Criteria for Indications at Tube Support Plates" Appendix A were utilized. This IPC provides a correlation between tube burst pressure and Bobbin Coil voltage amplitude. Tube support plate intersections examined by Bobbin Coil which had either a percent through wall or Distorted Signal Indication (DSI) indicative of Outer Diameter Stress Corrosion Cracking (ODSCC) or Intergranular Attack (IGA) were retested using MRPC. Tubes with MRPC confirmed degradation were plugged. SCE&G elected to pull the 3 tubes with the highest voltages (9.84, 11.59, and 22.32 volts) in order to obtain more information on the morphology of the degradation. SCE&G's activities were discussed with the NRC VCSNS Project Manager and other NRC personnel on several occasions.

On March 29, 1993, the eddy current examinations were completed. The results yielded an inspection Category C-3 determination since more than 1% of the tubes in the hot leg tubesheet region were defective. Prompt notification was made to the NRC Operations Center upon entry into Category C-3 for the steam generators. The notification was made per the requirements of Technical Specification Section 4.4.5.5.c and 10CFR50.72(b)(2)(i). Plugging of defective tubes was completed on April 9, 1993.

CAUSE OF EVENT:

The Category C-3 tube degradation is the result of Primary Water Stress Corrosion Cracking (PWSCC) in the hot leg tubesheet area.

Based on WCAP-13522, the degradation in the tube support plate area has been attributed to ODSCC. The results of the pulled tube analyses will be used to confirm this assessment.

ANALYSIS OF EVENT:

By performing tube plugging in accordance with Technical Specification requirements, defective tubes were removed from service, thereby ensuring that there is no reduction in the degree of protection provided to the public.

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Virgil C. Summer Nuclear Station	0 5 0 0 0 3 9 5	9 3	— 0 0 1 2 —	0 0	0 4 OF 0 4

TEXT (If more space is required, use additional NRC Form 366A's) (17)

CORRECTIVE ACTIONS:

All defective tubes were plugged in accordance with Technical Specification requirements. Subsequent inservice inspections will be performed as required by the VCSNS Technical Specifications.

In order to confirm the degradation mechanism in the tube support plate area, 3 tubes (1 intersection per tube) were removed from Steam Generator B and sent to Westinghouse for examination and testing. Preliminary results from the testing will be provided to the NRC as soon as they are available. Final documented results are expected by the end of September and a supplement to this LER will be provided by October 15, 1993. Any additional corrective actions will be specified in the supplement.