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May 6, 1977

Docket No.: 50-280

Virginia Electric & Power Company
ATTN: Mr. W. L. Proffitt
Senior Vice President - Power
P.O. Box 26666
Richmond, Virginia 23261

Gentlemen:

Enclosed is a signed original of an Order for Modification of License, dated May , 1977, issued by the Commission for the Surry Power Station Unit No. 1. This Order amends Facility Operating License No. DPR-32 permitting continued operation of Surry Unit No. 1 for six equivalent months of operation, beyond midnight May , 1977, and releates to the steam generator repair program license provisions of the NRC Order of February 8, 1977, and Corrective Order dated February 11, 1977. An Appendix A-1 to the license is being added in order to implement the restrictions of Ordered License Condition 3.E.(4) regarding reactor coolant activity.

In order to confirm the rate of denting with time, provide the probe data for those steam generator tubes which have been probed more than once with various sized probes during the recently completed inspection and previous inspections of Unit No. 1. Include the location in the tubes where probes would not pass. Provide your analysis and conclusions as to the rate of change of dent diameter based on these data.

A copy of the related Safety Evaluation is also enclosed. The Order is being filed with the Office of the Federal Register for publication.

Sincerely,

Original Signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Const. 2

Enclosures:
1. Order for Modification of License
2. Appendix A-1 to License No. DPR-32
3. Safety Evaluation

cc w/enclosures:
See next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE	AD-OR:DOR	AD-OT:DOR
SURNAME	*KRGoller	*DEisenhut
DATE	5/ /77	5/ /77

OFFICE	ORB#4:DOR	ORB#4:DOR	C-ORB#4:DOR	OELD	D:DOR	NRR
SURNAME	RIngram*	MFairtile	RReid	Grossman*	VStello*	EGCase
DATE	5/ /77	5/6/77	5/6/77	5/ /77	5/ /77	5/ /77

MEF

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Docket No.: 50-280

Virginia Electric & Power Company
ATTN: Mr. W. L. Proffitt
Senior Vice President - Power
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Richmond, Virginia 23261

Gentlemen:

Enclosed is a signed original of an Order for Modification of License, dated May , 1977, issued by the Commission for the Surry Power Station Unit No. 1. This Order amends Facility Operating License No. DPR-32 permitting continued operation of Surry Unit No. 1 for six equivalent months of operation, beyond midnight May , 1977, and relates to the steam generator repair program license provisions of the NRC Order of February 8, 1977, and Corrective Order dated February 11, 1977. An Appendix A-1 to the license is being added in order to implement the restrictions of Ordered License Condition 3.E.(4) regarding reactor coolant activity.

A copy of the related Safety Evaluation is also enclosed. The Order is being filed with the Office of the Federal Register for publication.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Order for Modification of License
2. Appendix A-1 to License No. DPR-32
3. Safety Evaluation

cc w/enclosures: See next page

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SURNAME	RIngram	MFairtile	RWReid	Grossman	VStello	EGCase
DATE	5/6/77	5/6/77	5/ /77	5/6/77	5/6/77	5/6/77

NRC Form 318 (2-76) NRCM 02040 ☆U.S. GOVERNMENT PRINTING OFFICE: 1976-634-782

Virginia Electric & Power Company

cc w/enclosure(s):
Michael W. Maupin, Esq.
Hunton, Williams, Gay & Gibson
P. O. Box 1535
Richmond, Virginia 23213

Swem Library
College of William & Mary
Williamsburg, Virginia 23185

Mr. Sherlock Holmes, Chairman
Board of Supervisors of Surry County
Surry County Courthouse
Surry, Virginia 23683

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S.W.
Washington, D.C. 20460

U. S. Environmental Protection Agency
Region III Office
ATTN: EIS COORDINATOR
Curtis Building (Sixth Floor)
6th and Walnut Streets
Philadelphia, Pennsylvania 19106

Commonwealth of Virginia
Council on the Environment
903 9th Street Office Building
Richmond, Virginia 23219

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of VIRGINIA ELECTRIC AND POWER COMPANY Surry Power Station, Unit No. 1	)))))	Docket No. 50-280

ORDER FOR MODIFICATION OF LICENSE

I.

Virginia Electric and Power Company (the Licensee), is the holder of Facility Operating License No. DPR-32 which authorizes the operation of the nuclear power reactor known as Surry Power Station, Unit No. 1 (the facility) at steady state reactor power levels not in excess of 2441 thermal megawatts (rated power). The reactor is a pressurized water reactor (PWR) located at the Licensee's site in Surry County, Virginia.

II.

On February 8, 1977, the staff issued an Order related to License No. DPR-32 which addressed operation of Surry Power Station Unit No. 1 under conditions in which steam generator tubes have been plugged as a result of the tube denting caused by corrosion of the tube support plate in the annular spaces between tube and the tube support plate. Subsequently on February 11, 1977, we issued a Safety Evaluation supporting the Order. In order to perform an inspection of the steam generators, the February 8, 1977 Order limited operation to 60 equivalent days. The licensee has since that time submitted inspection and repair programs dated April 15 and 29, 1977* and has shutdown to perform the inspection and repairs. The NRC staff has evaluated the results of the inspection and repair program and has assessed whether continued operation of the facility would be safe. This evaluation is set forth in the staff's concurrently issued Safety Evaluation relating to the steam generator tube integrity.

With respect to the effect of increased stress in the tube support plate as a result of tube support plate growth, the staff has concluded that neither buckling of the tube support plate nor damage to the steam generator shell through the wrapper and channel spacer would develop.

Continued growth of the tube support plate continues to impose stresses on the tubes and may result in the development of stress corrosion cracks in the tubes at denting locations. The staff has considered the effect of the development of stress corrosion cracking during the course of operation of this facility, and has assessed the potential effect of such cracks in conjunction with steam line break and loss of coolant accident events. The staff has concluded that under the additional limitations on tube leakage set forth in this Order, the potential

*Copies of (1) the Licensee's submittals dated April 15 and 29, 1977, (2) February 8, 1977 Order and Corrective Order dated February 11, 1977, relating to DPR-32, (3) this Order for Modification of License, In the Matter of Virginia Electric and Power Company, Surry Power Station, Unit No. 1, Docket No. 50-280, and (4) the Commission's concurrently issued Safety Evaluation supporting this Order are available for inspection in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555 and at the Swem Library, College of William and Mary, Williamsburg, Virginia. A copy of items (2), (3), and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555. Attention: Director, Division of Operating Reactors.

effect of continued denting on the consequences of the steam line break event would not exceed a fraction of Part 100, and any effects of continued denting on LOCA events would not be significant. These events are of extremely low probability, especially for the limited period covered by this Order. The additional limitations set forth in this Order will provide reasonable assurance that the public health and safety will not be endangered.

After discussion with the staff, the licensee proposed in his April 29, 1977, submittal to modify the limitations applicable to this facility. The NRC staff believes that under the circumstances, the limitations proposed by the licensee and the additional ones added by the NRC staff are appropriate and should be confirmed by NRC order.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS ORDERED THAT Facility Operating License No. DPR-32 is hereby amended by deleting the provisions of the NRC Order dated February 8, 1977, as corrected February 11, 1977, and replacing in its entirety existing paragraph 3.E. of the license with the following:

E. Steam Generator Inspection

(1) Unit No. 1 shall be brought to the cold shutdown condition in order to perform an inspection of the steam generators within six equivalent months of operation from May , 1977. Nuclear Regulatory Commission approval shall be obtained before resuming power operation following this inspection.

For the purpose of this requirement, equivalent operation is defined as operation with a primary coolant temperature greater than 350°F.

(2) Primary to secondary leakage through the steam generator tubes shall be limited to 0.3 gpm per steam generator, as described in the Safety Evaluation. With any steam generator tube leakage greater than this limit the reactor shall be brought to the cold shutdown condition within 24 hours and Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

(3) If leakage from two or more tubes in the plant in any 20-day period is observed or determined, the reactor shall be brought to the cold shutdown condition within 24 hours and Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation or if two leaks are observed after the reactor is in cold shutdown Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

(4) The concentration of radioiodine in the primary coolant shall be limited to 1 μCi/gram during normal operation and to 10 μCi/gram during power transients as defined in

the concurrently issued addition of Appendix A-1 to the Technical Specifications of the license.

FOR THE NUCLEAR REGULATORY COMMISSION

Edson G. Case

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Dated in Bethesda, Maryland
this 6th day of May 1977.

APPENDIX A-1 TO TECHNICAL SPECIFICATIONS

TO

FACILITY OPERATING LICENSE NO. DPR-32

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.1.D The specific activity of the primary coolant shall be limited to:

a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq$ $41/\overline{E}$ μCi/gram.

APPLICABILITY: MODES 1, 2, 3, 4 and 5

ACTION:

MODES 1, 2 and 3*

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 but less than 10. μCi/gram Dose Equivalent I-131 operation may continue for up to 48 hours provided that operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding 10. μCi/gram Dose Equivalent I-131 be in at least HOT STANDBY with T_{avg} < 500°F within 6 hours.

c. With the specific activity of the primary coolant > $41/\overline{E}$ μCi/gram, be in at least HOT STANDBY with T_{avg} < 500°F within 6 hours.

MODES 1, 2, 3, 4 and 5

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > $41/\overline{E}$ μCi/gram, perform the sampling and analysis requirements of item 4a of Table 3.1.D-1 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses together with the following information:

*With T_{avg} $\geq$ 500°F.

ACTION: (Continued)

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of de-gassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1.0 μCi/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table (3.1.D-1).

TABLE 3.1.D-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 or 41/$\bar{E}$ μCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 & 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

#Until the specific activity of the primary coolant system is restored within its limits.

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

PLANT SYSTEMS

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.6.C The specific activity of the secondary coolant system shall be $\leq$0.10 µCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the specific activity of the secondary coolant system > 0.10 µCi/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table (3.6.C-1).

TABLE 3.6.C-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (µCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

* Excluding decay heat.

** Reactor vessel head unbolted or removed and fuel in the vessel.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING ORDER FOR MODIFICATION OF LICENSE
VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNIT NO. 1
DOCKET NO. 50-280

INTRODUCTION

On February 8, 1977, we issued an Order for Modification of License No. DPR-32 to Virginia Electric and Power Company (VEPCO or the licensee) which authorized operation of Surry Power Station Unit No. 1 with plugged steam generator tubes; the Order limited operation to 60 equivalent days. Under conditions of that Order the licensee placed Unit No. 1 in cold shutdown on April 11, 1977, to perform required inspections and repairs. The licensee submitted his proposed inspection program by letter dated April 15, 1977 and in a letter dated April 29, 1977, submitted the proposed repair program. The April 29 letter requested our authorization to restart Surry Unit No. 1 and operate it for six equivalent months based on the results of the repair program submitted in the letter.

BACKGROUND

Water Chemistry

For many years a sodium phosphate treatment for PWR secondary coolant was widely used for U-tube design steam generators that removed precipitated or suspended solids by blowdown. It was successful as a scale inhibitor, however, in the early use, many PWR U-tubed steam generators with Inconel-600 tubing experienced stress corrosion cracking. The cracking was attributed to free caustic which can be formed when the Na/PO_4 ratio exceeds the recommended limit of 2.6. In addition, some of the insoluble metallic phosphates, formed by the reaction of sodium phosphates with the dissolved solids in the feedwater, were not adequately removed by blowdown. These precipitated phosphates tended to accumulate as sludge on the tube sheet and tube supports at the central portion of the tube bundle where restricted water flow and high heat flux occurs. Phosphate concentration (hideout) at crevices in areas of the steam generator, noted above, caused localized wastage resulting in thinning of the tube wall. The problem of stress corrosion cracking was corrected by maintaining the Na/PO_4 ratio between 2.6 and 2.3. Although the recommended Na/PO_4 ratio was maintained, it did not correct the phosphate hideout problem that caused wastage of the Inconel-600. Largely to correct the wastage and caustic stress corrosion cracking encountered with the phosphate treatment, most PWRs with a U-tube design steam generator using a phosphate treatment for the secondary coolant have now converted to an all volatile chemistry (AVT).

In 1975, radial deformation, or the so-called "denting," of steam generator tubes occurred in several PWR facilities after 4 to 14 months operation, following the conversion from a sodium phosphate treatment to an AVT chemistry for the steam generator secondary coolant. Tube denting occurs predominantly in rigid regions or so-called "hard spots" in the tube support plates. These hard spots are located in the tube lanes between the six rectangular flow slots in the support plates near the center of the tube bundle and around the peripherial locations of the support plate where the plate is wedged to the wrapper and shell. The hard spot areas do not contain the array of water circulation holes found elsewhere in the support plates.

The phenomenon of denting has been attributed to the accelerated corrosion of the carbon steel support plates at the tube/tube support plate intersection (annuli). The corrosion product (magnetite) from the carbon steel plate has expanded volumetrically to exert sufficient compressive forces to dent the tube and crack the tube support plate ligaments between the tube holes and water circulation holes, due to an in-plane expansion of the support plate. As a result of the tube support plate expansion, the rectangular flow slots began to "hourglass;" i.e., the central portion of the parallel flow slot walls have moved closer so that some of flow slots are now narrower in the center than at the ends.

U-Bend Cracks

On September 15, 1976, during normal operation, one U-tube in the innermost row parallel to the rectangular flow slots in steam generator A at Surry Unit No. 2 rapidly developed a substantial primary to secondary leak (about 80 gpm). After removal of the damaged tube and subsequent laboratory analysis, it was established that the leak resulted from an axial crack, approximately 4-1/4 inches in length, in the U-bend apex due to intergranular stress corrosion cracking that initiated from the primary side. Since the initial parallel flow slot wall in the top support plate has moved closer, the support plate material around the tubes nearest this central portion of these flow slots has also moved inward, in turn forcing an inward displacement of the legs of the U-bends at these locations. This inward movement of the legs of the U-bends at these locations caused an increase in the hoop strain and ovality of the tubes at the U-bend apex. It is this additional increase in strain at the apex of the U-bend which is believed to be required to initiate stress corrosion cracking of the Inconel 600 alloy tubing exposed to PWR primary coolant.

Laboratory examination of 71 U-bends removed from flow slot locations in rows 1, 2, and 3 of the Surry Units No. 1 and 2 and Turkey Point Unit No. 4 steam generators has shown that intergranular cracking at the U-bend apex was found only in the row 1 tubes.

Of the 71 tubes removed from these operating reactors, which are the most severely affected, no cracks have been found in tubes with computed equivalent strains less than 13.5% after approximately 11,065 hours of effective full power operation since detection of the first tube dent. However, this same equivalent operating time led to the tube failure at Surry Unit No. 2, where the equivalent strain was estimated to be >14.3%. This indicates a strain level at which rapid development of stress corrosion cracking may occur in U-bends of steam generators of this design.

Recent test work also indicates that long incubation periods are needed for the development of stress corrosion cracking at some strain rates.[2-5] Tests indicated that at 12.5% outer fiber strain,[1] Inconel 600 U-bend specimens tested in high purity water at 650°F took a long incubation time (>12,000 hours) for the nucleation of an intergranular crack, longer than 13,000 hours for >30% penetration and more than 18,000 hours to fail.

Although these test results are not directly applicable to the PWR steam generator tubing at Surry, they do confirm the observed operating experience that (1) a long incubation time is required to initiate intergranular cracking in Inconel 600 material, and (2) a high strain is required for crack propagation.

In this regard, the staff requested that the licensee address the following concern:

"Hourglassing" may continue and close the flow slots in the top support plate increasing the strain at the U-bend apex of the tubes in rows 2 and beyond.

In response to this concern, and to supplement plugging of row 1, VEPCO has installed stainless steel 304 alloy blocks in each of the six flow slots in the top support plate of all three Surry Unit No. 1 steam generators. These blocks prevent further closure of the flow slots and the inward displacement of the legs of the U-bends, thereby preventing further anticlastic straining at the U-bend apex of these tubes in rows 2 and beyond. As a result, intergranular stress corrosion cracking of those tubes at the U-bends in row 2 and beyond is not anticipated during near term (next year) normal operation. However, the flow slot blocking devices would cause: (1) an increase in strain in the support plate, (2) peripherial expansion of the support plate between wedge locations, (3) an increase of tube denting in the "hard spot" regions, and (4) additional bearing stresses on the wedges, wrapper, channel, and steam generator shell due to the peripherial expansion of the support plate. The net overall effect of flow slot blocking devices would be similar to complete closure of the flow slots. However, VEPCO had also increased selective tube plugging in the hard spot regions for the prevention of tube leaks at dented locations.

Support Plate Expansion

Continued growth of the magnetite in the tube-tube support plate annuli results in a non-uniform increase in strain in the support plates and corresponding in-plane expansion. In this regard, the staff requested that the licensee address the following concerns:

"1. Severe cracking of the support plate may result due to the continuing in-plane expansion of the support plate.
2. The rate of in-plane expansion in any support plate could increase the severity of tube denting in "hard spot" regions. Severe denting would restrain the tubes in the support plate and the plate may have a tendency to buckle or otherwise deform and thus exert additional bending loads on tubes.
3. With the closure of all the flow slots in any one support plate additional loads could be transmitted (due to the in-plane expansion of the plate) to the wedges, wrapper, channel spacer, tubes, and the steam generator vessel.
4. Thermal-hydraulic performance could be affected with the closure of all the flow slots in any support plate."

Anti-vibration Bar Fretting

On November 17, 1976 Southern California Edison Company (SCEC) reported to I&E, Region V, that, during the inspection of the San Onofre Unit No. 1 steam generators, excessive wear or mechanical fretting of anti-vibration bars was found in one of the steam generators. A failure of these bars could result in excessive flow induced vibration that might affect tube integrity, especially for those plants where the tube denting phenomenon was observed at the top support plate. Subsequent investigation revealed that the anti-vibration bar design of San Onofre Unit No. 1 and Connecticut Yankee is unique in comparison with other Westinghouse plants. Differences in the design are summarized as follows:

a. Materials - carbon steel for San Onofre Unit 1 and Connecticut Yankee; Inconel 600 for new models (44 and 51).
b. Bar Cross-section - 3/8 inch round bars; changed to square bars in the new models.
c. Clearances - (L-35 mils); was changed to (L-20 mils) for new models where L is the tube spacing.
d. Changes in V-bar configuration and spacing.

DISCUSSION

On February 8, 1977, Virginia Electric and Power Company (VEPCO or the licensee) was ordered to amend the Facility Operating License No. DPR-32 by adding provisions which require that, among other operational limitations, Surry Unit No. 1 be brought to cold shutdown condition to perform an inspection of the steam generators within 60 equivalent days of operation from February 8, 1977. In addition, the licensee was required to provide information delineated in Appendix A, to the Order, within 45 days of the date of the Order.

By letters dated March 23 (Westinghouse NS-CE 1385) and 25 (VEPCO Serial No. 031B/11977), 1977, the licensee provided information in response to the request stated in Appendix A to the Order. On April 15, 1977, VEPCO submitted by letter (Serial No. 057/021177) an outline of the surveillance program for the current outage as dictated by the Order.

On April 29, 1977, the licensee, by letter (Serial No. 057A/021177), submitted the results of the inspection of the Unit 1 steam generators and support plate analyses and the plugging criteria for dented tubes that were implemented, and proposed to resume power operation for six (6) equivalent months.

Surveillance Program

The steam generator surveillance program conducted during the current outage consists of the following:

(1) An ECT examination program was performed from the cold leg side over the U-bends for all unplugged tubes located in rows 1 to 5.

(2) Tube ID gauging and accompanying ECT was performed from the hot leg side for tubes in the interior tube lane regions, the wedged region and the patch plate region for all three Unit 1 steam generators. For the ID tube gauging for denting, three different size probes were used; i.e., 0650, 0610, and 0.540 inch probes. The accompanying ECT used different frequencies to detect any tube flaw. Finally, any tubes which will not allow passage of the 0.610 probe will be probed by the 0.540 probe at 25 kH_z for sludge mapping.

(3) A photographic and visual inspection program was conducted for all three steam generators through the lower handholes and the 3-inch inspection port in one steam generator to inspect the conditions of flow slots in the lower and top tube support plates.

(4) The wrapper to shell annulus gauging was performed to determine the support plate deformation at the outer edges.

Results of Inspection and Corrective Actions

The results of the inspection did not reveal any unusual phenomenon or unexpected results for the type of degradation previously observed and documented. The U-bend ECT of rows 1-5 in each steam generator did not discover any discernible flaw indications. However, seven (7) row 2 tubes and two (2) row 3 tubes did not permit passage of the 0.540 inch probe through the U-bend because of the small bend radius and some ovality. These tubes were plugged as a preventive measure.

The inspection of the flow slots in the top support plate through the 3-inch inspection port revealed that the flow slot blocking devices were intact, and no detrimental conditions were noted. Through the lower handholes, the conditions of the lowest tube support plates in all three steam generators were similar to that discovered in Unit No. 2; i.e., cracks were found in fifteen (15) out of a total of eighteen (18) flow slots and three flow slots in S/G-1A were completely closed.

The shell to wrapper annulus measurements were made in S/G-B and C. These data will be used to correlate the actual and the predicted support plate expansion. No detrimental conditions were noted during the inspection.

The combined tube ID gauging and ECT of dented tubes did not find any flaw indications. Severely dented tubes were plugged in accordance with the tube plugging criteria stated below:

Plugging Criteria

1. All tubes which didnot pass the 0.540 inch probe were plugged.

2. Additionally, for six (6) months operation, three (3) tubes beyond (i.e. higher row numbers) any tube in columns 15 to 80 which did not pass the 0.540 inch probe were plugged; for such tubes in columns 1 to 14 and 81 to 94 four (4) tubes beyond were plugged.

3. All tubes which did not pass the 0.610 inch probe were plugged.

4. The tubes in any column for which plugging under criteria (a), (b) or (c) above was implemented were also plugged in the lower row numbered tubes back to the tube lane if not already plugged.

5. As a conservative measure, tubes completely surrounding all known leaking tubes -- including the diagonally adjacent tube were plugged, if not already covered by the foregoing criteria.

6. Additional preventive plugging was implemented for tubes R46C41 and R46C54 in each steam generator, even though no eddy current indications were reported for these tubes. This action was taken to alleviate concern for those unique tube locations:

a. At the perimeter of the plate

b. Adjacent to a discontinuity (cut) in the plate created by the installation of the patch plate

c. Adjacent to plug welds in circulation holes.

Criteria stated in (2) for plugging the tubes beyond those which do not allow the passage of a 0.540 inch probe are based on the movement of the region of tubes which are severely dented. This region is bounded by the plate strain contour that corresponds to a 14% hoop strain in the tubes. Based on the history of previous leak locations with the exclusion of the patch plate leaks, the growth of the plate strain intensity contour corresponding to the 14% hoop strain in tubes is estimated to be about one half of a tube row per month in the middle tube lane area and about two thirds of a tube row in regions near the outer edges of the tube lane. The 14% hoop strain in tubes appears to be bounding for all leaking tubes observed in both Surry Units 1 and 2.

The above discussed tube plugging criteria implemented on Surry Unit 1 combined with previous plugging resulted in 18.5% of the steam generator tubes being plugged (as compared to 16.4% for Unit 2).

"Islanding" Effect

With respect to the possible loss of lateral support on some inner row tubes due to the so-called "islanding"; i.e., broken support plate pieces moving into flow slots, the concern may be alleviated by the fact that most inner row tubes (first three rows) will be plugged. The licensee has also submitted results of an analysis of fluid structural vibrations considering loss of one, two, and three lateral supports. The maximum vibration amplitude was calculated to slightly exceed one half of the gap between adjacent tubes for the case when three lateral supports were lost. This resulted in a slight increase in bending stress in the tube.

Potential for U-Bend Cracking

Prior to February 8, 1977, the licensee had installed stainless steel blocks in each of the flow slots in the top tube support plate of all three Surry Unit 1 steam generators. These blocks prevent further closure of the flow slots and the inward displacement of the legs of the U-bends, thereby preventing further anticlastic straining at the U-bend apex of those tubes in rows 2 and beyond. With the flow slot blocking devices, the susceptibility for intergranular cracking of tubes beyond row 1 is substantially less because of the larger U-bend radius, less plastic pre-straining, and smaller residual stresses.

As a part of the submittal on March 25, 1977, the licensee provided additional analysis to show that indeed the residual hoop and axial stresses for tubes in rows 2 and beyond decrease as the U-bend radius increases. The magnitude of the residual stresses are dependent on the degree of cold working of the tube cross-section during the forming of the small radius U-bend. Bending of small radius U-bend tubes causes their cross-section to ovalize. However, the presence of an internal ball mandrel during bending of tubes for rows 1 and 2 forces the cross-section back into a configuration with considerably less ovality. Upon removal of the ball, the cross-section rebounds elastically and the residual hoop stresses on the inside surface of the tube are tensile at the top and bottom positions of the U-bend apex. Bending of tubes for rows greater than 2 does not employ the ball mandrel. Hence those tubes adopt the characteristic oval shape. After bending, the tube cross-section rebounds elastically and the corresponding residual hoop stress for tubes in row 3 and beyond are compressive. The degree of residual tube ovality is a function of the bend radius, with the greatest ovality associated with the tubes with the smallest bend radii, i.e., row 3.

The analyses indicate that the residual hoop stresses at the U-bend apex are equal to one half of the yield strength, or about 26,000 psi and the residual axial stress in small radius U-bend tubes is 15000 psi. The magnitudes of these residual stresses are less than the ASME Code minimum yield strength. Therefore, with these low residual stresses at the U-bend apex and the fact that additional stresses are prevented by the flow slot blocking devices in the top support plate, the potential for stress assisted intergranular attack at the U-bend for tubes in rows 2 and beyond is minimal.

EVALUATION

By letter dated April 29, 1977, the licensee proposed to continue power operation of Surry Unit No. 1 for an additional six (6) months beyond the sixty (60) days equivalent power operation approved by NRC on February 8, 1977. This proposal was based upon the results of the extensive examination program and the supporting conclusions discussed above. The NRC staff has reviewed the information submitted by the licensee and concurs in the following:

1. Further U-bend failures are not likely to occur for near term continued operation because of the following:
   a. Laboratory examinations of 71 tubes removed from Surry Units No. 1 and 2 and Turkey Point Unit No. 4 steam generators indicate that cracking was confined only to row one tubes.
   b. All the tubes in row one and most tubes in row two are plugged.
   c. Continued "hourglassing" of the top support plate flow slots was prevented by the installation of flow slot blocking

devices thereby preventing further inward displacement of the U-bend legs and anticlastic straining at the U-bend apex of those tubes in rows 2 and beyond.

d. The susceptibility for U-bend cracking of tubes in row 2 would be substantially less because the residual hoop stress at the U-bend apex is 25-35% lower than the ASME Code minimum yield strength for Inconel 600 alloy tubing and the effective U-bend strain is 30-50% lower than in row 1.

e. U-bend cracking in rows 3 and beyond will not occur because the residual hoop stresses on the inside surface of the tube are compressive at the top and bottom positions of the U-bend apex, and thus the potential for stress corrosion cracking is not possible.

2. Support plate expansion or continuing magnetite growth in the proposed period of operation will have insignificant effects on the wrapper and the steam generator vessel. Therefore, the wrapper and the vessel integrity during normal operating and accident conditions will not be affected by continued support plate expansion.

Due to the blocking of the flow slots in the top support plates and the possible closure of flow slots in lower support plates, additional loads could be transmitted to the steam generator shell through the load path of the support plate, wedge, wrapper and channel spacer. Based on preliminary "crush" tests performed by Westinghouse the maximum load that can be developed along this load path is 60,000 pounds.

Analysis of the bearing stress along this path indicates that all stresses are less than the yield strength. Such stresses on the steam generator shell are highly localized and self limiting and will not adversely affect the integrity of the shell under accident conditions.

3. With respect to the effect of continued magnetite growth that causes the support plate expansion and thus denting, the preventive plugging program that was implemented in accordance with the criteria discussed previously in the section of this evaluation entitled Plugging Criteria, is adequate. In this regard, the NRC staff also considered the following additional supportive reasons:

   a. Refined analyses of the support plate expansion (to complete flow slot closure) indicated small hard spot strain increases and plate perimeter deformations.

   b. All of the tubes in hard spot regions and those that have leaked previously have been plugged, based on the criteria derived from past experience and the correlation between the strain predictions and field gauging results. This corresponds, in general, with the calculated strain pattern due to a conservatively estimated magnetite growth rate.

   c. All leaks associated with dented tubes experienced to date have been small, well below commonly acceptable leakage limits.

d. Observed through-wall cracks in the dented regions; i.e., tube/support plate intersections, are constrained by the support plates; therefore, cracks should not burst during postulated accidents, unless the crack extends substantially beyond the tube support plate region.

e. Through wall cracks at dented locations, with the amount of leakages experienced to date, have been stable during normal operation (no rapid failures), and are not anticipated to become unstable during postulated accidents.

f. Even though some non-through-wall cracks may exist and may crack through during postulated accidents, the associated leakage rate with such an event would be similar to that resulting from through wall cracks found during normal operation and the crack would not be unstable. The licensee has plugged tubes based on a projection of the plate strain and associated tube strain (denting) for an additional six months of operation. This consideration is consistent with the rationale upon which the preventive plugging limits were set for wastage or fretting type of degradation.

g. Even if a LOCA or a MSLB were to occur during the proposed period of operation and some tubes were in a state of incipient failure, the consequence of such an event will not be severe as discussed in the Safety Evaluation issued February 11, 1977, in support of the February 8, 1977 Order, pages 12 through 15.

4. The fact that most of inner row tubes are plugged lessens the concern over the possible loss of lateral support of tubes due to the so-called "islanding" effect.

5. With respect to the tubes in the patch plate regions, no discernible flaws were found. As a preventive measure, tubes adjacent to plug welds in circulation holes were plugged. Therefore, the concern over the possible "new" hardspots around the patch plates is alleviated.

Operational Limitations

Because of the need to assure that any stress corrosion cracking which occurs during operation remains small and stable and that an extensive number of tubes do not incur penetrating cracks or substantial part through-wall cracks, the staff has developed certain additional operating limitations. These limitations are designed to assure the detection of the onset of tube degradation before it becomes a significant concern.

A limit for primary to secondary leakage of 0.3 GPM will assure that no individual cracks will reach such proportions that it may become unstable during normal or accident loading conditions.

A substantial increase in the frequency at which leaking tubes are encountered could signal the development of more extensive general degradation. The potential for such a development during operation has been substantially alleviated by the limitations described below, requiring operation to be terminated in the event that the frequency of the detection of leaking tubes per plant should increase substantially to more than 1 in twenty days. Specifically, the restriction is that operation is to be terminated if two (2) or more tube leaks per plant occur during any twenty (20) day period. This restriction limits the potential number of heat up and cool down cycles resulting from tube plugging and minimizes concern for possible thermal ratcheting.

At the end of the proposed six (6) month operating period, the unit shall be brought to cold shutdown condition for a reinspection of the conditions of the steam generators and to reassess the subsequent duration and mode of operation. The inspection program should be, as a minimum, as comprehensive as the one performed during the current outage. Detailed requirements will be determined by the NRC staff on the basis of the continuing operating experience.

CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this Order will not be inimical to the common defense and security or to the health and safety of the public. However, this conclusion is only applicable for six (6) months operation.

Date: May 6, 1977

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