October 8, 1977

Docket No.: 50-281

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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 October 8, 1977, issued by the Commission for the Surry Power Station Unit No. 2. This Order amends Facility Operating License No. DPR-37 permitting continued operation of Surry Unit No. 2 for six equivalent months of operation beyond October 8, 1977, and relates to the steam generator repair program license condition of the NRC Orders for Modification of License dated April 1, 1977 and August 17, 1977. Appendix A-1 to the license, issued April 1, 1977, is being continued 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. Safety Evaluation

cc w/enclosures: See next page

Check w/ OT for Comm

with comments provided to MF 10/8/77 DONE MBF

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MBF

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Virginia Electric & Power Company

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Mr. Sherlock Holmes, Chairman
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Commonwealth of Virginia
Council on the Environment
903 9th Street Office Building
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of VIRGINIA ELECTRIC & POWER COMPANY Surry Power Station, Unit No. 2	)))))	Docket No. 50-281

ORDER FOR MODIFICATION OF LICENSE

I.

Virginia Electric and Power Company (the licensee), is the holder of Facility Operating License No. DPR-37 which authorizes the operation of the nuclear power reactor known as Surry Power Station, Unit No. 2 (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 April 1, 1977, the staff issued an Order for Modification of License No. DPR-37 which addressed operation of Surry Power Station Unit No. 2 under conditions in which steam generator tubes have been plugged as a result of tube denting caused by corrosion of the tube support plate in the annular spaces between tube and the tube support plate. On August 17, 1977, an Order was issued to permit continued operation of

Unit No. 2 to September 15, 1977, under the conditions of the April 1, 1977 Order. The licensee was required to perform an inspection after the September 15 shutdown. The licensee's fuel cycle for Surry Unit No. 2 ended before September 15, 1977, and during the resulting shutdown the licensee performed the required inspection and plugged 180 additional tubes. Twenty-one tubes were plugged because of wastage degradation; the remaining 159 tubes were plugged following the denting plugging criteria given in the licensee's September 30, 1977 submittal. The data derived from the inspection demonstrates that the denting has followed the pattern predicted by the preventive plugging criteria established by the licensee with the staff. The additional plugging performed as a result of the inspection using the preventive plugging criteria will provide adequate steam generator integrity under the conditions of this Order for continued operation for an additional six month period. Four tubes in generator C were inadvertently not plugged. The staff has evaluated the potential effect that may be associated with these susceptible tubes. With the stringent limits under which this facility has been operating, leakage through these tubes would be detectable and the crack, if any, would be stable. Nevertheless, if the generator is inspected for any reason during the next six months, those four tubes are to be plugged, in addition to any other required corrective action. The NRC staff has evaluated the results of this inspection and repair program and has assessed continued safe operation of the facility. This evaluation is set forth in the staff's concurrently issued Safety Evaluation relating to steam generator tube integrity.

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 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 effect of such cracks in conjunction with the steam line break and loss of coolant accident events. The staff has concluded that under the limitations on tube leakage set forth in this Order, the effect of continued denting on LOCA events or on the consequences of the steam line break event would continue to be within those considered in connection with the April 1, 1977 Order. The limitations set forth in this Order will provide reasonable assurance that the public health and safety will not be endangered.

The licensee has proposed in his September 30, 1977 and October 6, 1977 submittals and after discussions with the NRC staff to continue the limitations applicable to this facility in the manner set forth in this Order. The NRC staff believes that the licensee's actions, under the circumstances are appropriate and should be confirmed by NRC Order.

Copies of the following documents are available for public 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, (1) licensee's submittals of September 30, 1977 and October 6, 1977, (2) Orders for Modification of License dated April 1 and August 17, 1977, (3) this Order for Modification of License, In the Matter of Virginia Electric and Power Company, Surry Power Station, Unit No. 2, Docket No. 50-281, and (4) the Commission's concurrently issued Safety Evaluation supporting this 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-37 is hereby amended by replacing in its entirety existing paragraph 3.E. of the license with the following:

E. Steam Generator Inspection

(1) Unit No. 2 shall be brought to the cold shutdown condition in order to perform an inspection of the steam generators within six months of equivalent operation from October 8, 1977.
Nuclear Regulatory Commission approval shall be obtained before resuming power operation following this inspection.

Equivalent operation is defined as operation with the reactor coolant at or above 350°F.

*/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.

(2) Primary coolant leakage from the primary system to the secondary system through the steam generator tubes shall be limited to 0.3 gpm per steam generator, as described in the NRC Safety Evaluation of April 1, 1977. With any steam generator tube leakage greater than this limit the reactor shall be brought to the cold shutdown condition within 24 hours. Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

(3) Reactor operation will be terminated if primary to secondary leakage which is attributable to 2 or more tubes occurs during a 20 day period. 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 Appendix A-1 to the Technical Specifications of the license. Appendix A-1 was issued with the April 1, 1977 Order and shall remain in effect for six equivalent months from October 8, 1977.

(5) Should the steam generators require an inspection prior to the expiration of the six month equivalent operating period, the following four tubes R33C75, R33C77, R34C73, and R38C73 in steam generator 2C shall be plugged.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Dated in Bethesda, Maryland
this 8th day of October 1977.

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. 2

DOCKET NO. 50-281

INTRODUCTION

By letter dated September 30, 1977, Virginia Electric and Power Company (VEPCO) requested NRC's approval for operation of Surry Unit No. 2 for six (6) equivalent full power months (EFPM) of operation. Surry Unit 2 has been operating under an NRC Order for Modification of License dated August 17, 1977, which permitted operation of the unit to September 15, 1977. The August 17 Order was a continuation of an Order dated April 1, 1977. Both Orders were issued in relation to steam generator problems that Surry Unit 2 has experienced and continues to experience.

Among other operational limitations, the NRC Orders required that during the September refueling outage all three steam generators in Unit 2 be inspected and that NRC approval be obtained prior to resumption of power operation.

DISCUSSION

Inspection Program

During the current shutdown, VEPCO carefully assessed the condition of all three steam generators in Surry Unit 2 to determine, to the extent possible, the causes for the continuing occurrences of leakage at the facility in order to evaluate whether the unit may continue to operate in a safe manner.

-Details of the steam generator inspection program actually conducted are summarized as follows:

1. Eddy current inspections were made in all three steam generators of the following tubes including all lower row numbered tubes back to the tube lane: R7C1, R12ca, R15C3, R17C4 through R17C11, R13C11 through R13C85, R17C85 through R17C91, R15C92, R12C93, and R7C94. In addition, an inspection was made of all tubes two rows beyond any tube that would not pass a 0.650 inch probe.

2. Eddy current inspections were made in all three generators in the one o'clock and eleven o'clock wedge regions including two rows beyond any tube restricting a 0.610 inch probe.

3. Eddy current inspections were made in all three generators in the patch plate regions including two rows beyond any tube not allowing passage of a 0.650 inch probe.

4. Tube gauging was conducted on all three steam generators utilizing three different size eddy current probes, i.e., 0.540, 0.610, and 0.650 inch probe diameters.

5. A complete Regulatory Guide 1.83 inspection program was performed in each of the three steam generators.

6. The first five rows of tubes in each of the three generators were eddy current inspected from the cold leg side through the U-bends.

7. Wrapper to shell annulus measurements were made on steam generator A.

8. An inspection of the seventh tube support plate flow slots through the 3 inch plug was done in steam generator A.

9. The flow slots in the first tube support plate were inspected in all three steam generators.

Results of Inspections and Corrective Actions

At the time of shutdown indicated leakage was not detectable in steam generator A, 0.07 GPM in B and 0.09 GPM in C. The initial inspection, at a 500 psig secondary side pressure, indicated no leaks in generator A. Leaks were found in tube R37C76 hot leg and at the plug in hot leg tube R14C62 in generator B and in hot leg tubes R46C48 and R37C76 and at the plugs in cold leg tubes R9C7 and R2C61 in generator C.

The leaking tubes were either in the wedge region or in the patch plate area. Both areas were included in the inspection just completed. The leaking plugs were repair welded using the Westinghouse technique.

Results of the eddy current inspection program just completed show that no tube leaks occurred within the tube lane inspection pattern used during the last outage. During the last 4.5 EFPM of operation 22 tubes in the tube lane regions became restricted to the point of preventing the passage of a .540 inch probe. All these tubes have been plugged and it was noted that all of these tubes are directly adjacent to the tubes preventatively plugged in March 1977. Also, after 4.5 EFPM of operation, all of the tubes in the tube lane region that allowed passage of the .540 inch probe but not the .610 inch probe were found to lie quite close to tubes preventatively plugged in March 1977. All of these tubes have now been plugged.

The inspection results indicate that in no case where there is a so-called "spiked" preventative plugging pattern near the tube lane has there been any significant restriction of tubes in the adjacent columns, or any tendency to "pyramid" over this past operating period.

In the eleven o'clock wedge regions 8 tubes did not allow passage of the .540 inch probe and 1 tube restricted the .610 inch probe. All of these tubes have been plugged. However, VEPCO has inadvertently failed to meet the plugging criteria in this area. Steam generator C tubes R33C75, R33C77, R34C73, and R38C73, which are all diagonally located next to tubes that would not pass a 0.540 inch probe, were not plugged. It was noted, however, that all of these tubes passed 0.610 inch probe and the first two passed a 0.650 inch probe. Also, the projected rates of strain growth in this area will result in less than 15% strain at which, VEPCO states, tube leakage has occured in the past. No tubes in the one o'clock wedge regions not previously plugged impeded the passage of any size eddy current probe.

In the patch plate areas 2 tubes did not allow passage of the .540 inch probe and both tubes have been plugged.

The first five rows of tubes were eddy current inspected from the cold leg side through the U-bends. No tubes with greater than the minimum identifiable level of 20% wall penetration were found.

With respect to the annulus measurement and support plate flow slot inspections no unusual indications were found. The data collected provided representative information to be used in the future for correlating predicted and actual support plate expansions.

Eddy current inspections in accordance with Regulatory Guide 1.83 were conducted in greater than 12% of the tubes in each steam generator. Any tube with an indication of greater than 40% wall penetration was plugged. This amounted to 6 tubes in steam generator A, 13 tubes in B and 2 tubes in C. This is a significant reduction compared to the number of tubes plugged because of wastage during the May 1975 and May 1976 inspections. In steam generator A 20 tubes were plugged in May 1976 and 35 in May 1975. In generator B 35 tubes were plugged in May 1976 and 29 in May 1975. In generator C 20 tubes were plugged in May 1976 and 68 tubes in May 1975.

The ID gauging for denting was performed utilizing a series of different probe sizes; i.e., 0.540, 0.610, and 0.650 inch in probe diameter. The areas probed were chosen on the bases of the analysis of the critical strain contours in the tube support plate and the predicted growth of magnetite at tube/tube support plate annulus. The tubes were initially gauged with the 0.650 probe. Those tubes that did not allow passage of this probe were then gauged with the 0.610 inch probe. Any tubes which did not allow passage of the 0.610 inch probe were then gauged with the 0.540 inch probe. The results of this gauging process did indicate a fair correlation with the strain analysis in the support plate. Severely dented tubes were plugged in accordance with the tube plugging criteria stated below.

Plugging Criteria

The licensee has implemented the following plugging criteria to justify a period of six (6) months operation:

a) All tubes which do not pass the 0.540 inch probe will be plugged.
b) Additionally, for in excess of six months operation, two tubes beyond (i.e., higher row numbers) any tube in columns 15-79 which does not pass the 0.540 inch probe will be plugged; for such tubes in columns 1-14 and 80-94 six tubes beyond will be plugged. Restricted tubes in the extreme corners of the tube lane (columns 2-4 and 92-94) are evaluated individually since the rates of growth of contours in the area are lower than the general rate of growth of the contours in the corner areas.

c) All tubes which do not pass the 0.610 inch probe will be plugged.

d) The tubes in any column for which plugging under criteria a, b, or c above is implemented will also be plugged in the lower row numbered tubes back to the tube lane, if not already plugged; unless the required plugging would connect two apparently unrelated areas.

e) As a conservative measure, tubes completely surrounding any known leaky tubes including the diagonally adjacent tube will be plugged, if not already covered by the foregoing criteria.

f) Additional preventive plugging will be implemented in the patch plate region. This plugging will include all tubes that:

   (1) restrict the 0.540 inch probe
   (2) restrict the 0.610 inch probe
   (3) surround leakers and tubes that restrict the 0.540 inch probe including the diagonally adjacent tube.

(g) Additional preventive plugging will be implemented at the wedge locations. This plugging will include all tubes that:

   (1) restrict the 0.540 inch probe
   (2) restrict the 0.610 inch probe
   (3) surround leakers and tubes that restrict the 0.540 inch probe including the diagonally next tube.

Criteria stated in (b) for plugging the tubes beyond those which do not allow the passage of 0.540 inch probe is based on the change in the region of tubes which are severely dented. This region is bounded by the plate strain contour that corresponds to a 12.5% 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 12% to 16% hoop strain range (i.e., the range in which over 90% of the leakers have occurred) is conservatively estimated to be about one third of a tube row per month in columns 15-79 and 1 tube row per month in columns 1-14 and 80-94.

EVALUATION

By letter dated September 30, 1977, VEPCO proposed to start Cycle 4 operation of Surry Unit No. 2 for a period of six (6) months. This proposal was based upon the results of the extensive examination program and the supporting conclusions discussed above. We have reviewed the information submitted by the licensee and our evaluation is as follows:

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 Unit Nos. 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 rows one and two and most of the tubes in row three are plugged.
   c. U-bend cracking in rows 3 and beyond will not occur because the residual hoop stress 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 closure 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 which causes the support plate expansion and thus denting, the preventive plugging program that was implemented in accordance with the criteria discussed previously is adequate. In this regard, we also considered the following additional supportive reasons:

   a. Refined strain analyses of the support plate expansion (to complete flow slot closure) indicated small strain or deformation increases in hard spot regions around the flow slots and plate perimeter.

   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.

d. Observed through-wall cracks in the dented regions, i.e., tube/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 (less than 0.3 GPM), have been stable during normal operation (no rapid failures), and are not anticipated to become unstable during postulated accidents.

f. Should some non-through-wall cracks exist and crack through during postulated accidents, the associated leakage rate with such an event would be similar to that resulting from a through wall crack found during normal operation and the crack would not be unstable. This consideration is consistent with the rationale upon which the preventive plugging limits were set for wastage or fretting type of degradation.

g. Should a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB) occur during the proposed period of operation and some tubes were in a state of incipient failure, the consequence of such an event would not be as severe as discussed in the Safety Evaluation included in the April 1, 1977 Order, pages 23 through 26.

4. As predicted, the residual phosphates in the sludge has caused continued wastage, but at a reduced rate, on the OD surfaces of tubes near the top of the tube sheet. The number of defective tubes (i.e., those degradations that exceeded the plugging limit) has been substantially reduced, in comparison with the previous inspections confirming that the extent of tube wastage is diminishing.

5. In the eleven o'clock wedge region in steam generator C, plugging criteria (g) was not satisfied. At this time the staff has seen no data to substantiate VEPCO's statement that tube leakage will not occur at strain levels less than 15%. However, the staff feels that the evaluations 3.(d), (e), (f), and (g) above adequately address any concern in this area. Additionally, Operational Limitation 4, has been imposed upon the licensee.

Operational Limitations

1. 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. If this limit is reached operation shall be terminated.

2. 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.

3. 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 steam generators and to reassess the subsequent duration and mode of operation. Detailed inspection requirements will be provided to the NRC staff prior to the conduct of the inspection.

4. If for any reason during the next six (6) months of EFPM of operation any of the steam generators must be reopened, i.e., exceed operational limitations 1 or 2 above, the 4 tubes inadvertently not plugged, and in violation of the licensee's plugging criteria, shall be plugged.

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 regulation 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) equivalent months operation.

Dated: October 8, 1977