February 11, 1977

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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:

On February 8, 1977, we issued an Order for Modification of License for Surry Power Station Unit No. 1. Our Order amended Facility Operating License No. DPR-32 and permits continued operation for 60 equivalent days from February 8, 1977. Our Order also contained other limitations for operation of Surry Unit No. 1. In our Order we indicated that our Safety Evaluation was in preparation and would be issued subsequent to the Order.

Enclosed is our Safety Evaluation relating to steam generator tube integrity upon which our Order of February 8, 1977, is based. Please note that the Safety Evaluation contains descriptive statements and implementing requirements regarding the limitations listed in our Order of February 8, 1977.

The February 8, Order contained an error in Section III, Paragraph 4, line 3. Enclosed is a signed original of a Corrective Order. A copy of the Corrective 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:
Safety Evaluation
Corrective Order
cc w/enclosure: See next page

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SAFETY EVALUATION REPORT
SURRY UNIT NO. 1 STEAM GENERATORS
DOCKET NO. 50-280

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 predominately 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 spots 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 Nos. 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 time 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 will prevent further closure of the flow slots and 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, channal 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 January 19, 1977, Virginia Electric & Power Company (VEPCO), the licensee, was issued Amendment 29 to Facility Operating License No. DPR-32 to operate Surry Power Station Unit No.1 twenty (20) equivalent days with a primary coolant temperature greater than 350°F. Prior to January 19, 1977, the licensee (VEPCO) had, by letters dated January 3, 1977 and January 14, 1977 submitted an analysis of steam generator tube integrity for Surry No. 1. This information expanded upon the previous analyses concerning the U-bend cracking phenomenon in the steam generators of Surry Unit No. 2. VEPCO has performed corrective action including the installation of flow slot blocking devices to prevent further occurrences of U-bend cracking in the steam generators of Unit No. 1 and proposed to return to power for two (2) effective full power months (EFPM). Because of staff's concern as to the adequacy of analyses of the effectiveness and the consequences of the proposed corrective actions described in the January 14, 1977 submittal, the proposed two months of power operation was not granted. Instead, a period of 20 EFPD's of operation was approved with a stipulation that additional information be provided, as delineated in the Appendix A to the January 19, 1977 license actions (Amendment 29). The response to these questions were transmitted by letter, dated February 4, 1977, in which the licensee had also requested an additional period of four (4) months of power operation beyond the current 20 EFPD operations granted on January 19, 1977.

In the January 14, 1977 submittal, the licensee provided calculations to indicate that an equivalent strain of 13.5% at the U-bend apex represents a lower bound for intergranular cracking as a result of tube support plate deformation at the center of the flow slot locations. All the row 1 tubes with I.D. cracks had equivalent strains of 13.6% to 15.2% and those tubes with less than 13.5% had no cracks. The

total effective strain at the U-bend apex for all row 1 tubes would be approximately 15.7% if complete flow slot closure were to occur.

Similar calculations for tubes in rows 2, 3 and 4 indicate that the total equivalent strain decreases as the U-bend radius increased, even if the flow slots were to close completely. The maximum equivalent strain for any tube in row 2 is 10.1% to 10.9%, 7.4% to 8.3% for any tube in row 3, and 6.3% to 7.1% for any tube in row 4. Therefore, the susceptability for intergranular cracking of tubes beyond row 1 would be substantially less because of the larger U-bend radius,less plastic pre-straining, and thus small residual stresses. As a further corrective action to prevent the possibility of intergranular cracking at the U-bend apex for tubes beyond row 1, the licensee (VEPCO) has installed stainless steel blocksin each of the flow slots in the top tube support plate of all three Surry Unit 1 steam generators. These blocks will prevent further closure of the flow slots and 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.

The licensee had also provided results of preliminary calculations to evaluate the effects of installing flow slot blocking devices. First, the calculations were made for three loading conditions corresponding to the uniform in-plane growth of the top support plate at 0.014 and 0.021 inch per inch strains, and an uneven in-plane growth of the top support plate at 0.042 and 0.030 inch per inch strains in the hot and cold sides, respectively. Results of these three cases are quite similar all indicating maximum strains in the hard spot regions. There are no significant changes in the regions of high strain in going from 0.014 to 0.042 inch per inch strain. These three analyses were performed without the flow slot blocking devices and, therefore, represent the in-plane growth of the top support plate during past operation.

In order to evaluate the effects of the flow slot blocking, further analyses were made for a loading condition, corresponding to the 0.021 inch per inch strain, applied to the already deformed plate that had been subjected to the loading condition corresponding to 0.014 inch per inch strain, with and without the blocking devices. Results of these two analyses indicated a negligible amount of increase in strain and a small amount of deformation of the perimeter of the plate. Therefore, the licensee concluded that:

1. The expansion of the periphery is not in direct proportion with the increased expansion and,
2. the expansion of the plate at the perimeter is not worse with the flow slots blocked than it is without the blocks inserted.

In the February 4, 1977 submittal, the licensee correlated support plate expansion with actual months of operation by means of a finite element model which utilized a pseudo-thermal expansion technique. In order to do this, relationships between field data, EFPM's and results of the finite element analysis were established. Since the denting phenomenon extends over the entire plate, there is good correlation between measured denting and expansion of a dented plate. Although the finite element model is not detailed enough to yield denting rates, it does quantify the extent of flow slot closure for a prescribed expansion. Since the amount of closure over an extended period of EFPM's is available from field data, a relationship between model closure and EFPM's can be established. The rate of expansion is independent of boundary effects, the insertion of blocking devices, and time. The procedure for calculating the rate of expansion per EFPM is as follows:

For a plate expansion of 0.014 inch per inch in the hot side and 0.010 inch per inch in the cold side, the resulting average flow slot closure is 0.675 inch. For the actual plate, the most conservative

rate of closure as derived from field data for the top support plate is 0.15 inch/month. Thus, the 0.675 inch of closure represents 4.5 months of power operation, and the equivalent strain rates or the magnetitie growth rates are determined to be 0.0031/0.0022 inch per inch per month for the hot/cold side. Therefore, for an additional two months of operation, additional loads equivalent to 0.0062/0.0044 inch per inch strains in the hot/cold side were applied. Similarly, an equivalent loading condition with 0.0124/0.0088 inch per inch strains in the hot/cold side represents an additional four months of operation beyond the insertion of blocking devices in all six flow slots in the top support plate. The results of these two analyses show an insignificant change in the strain pattern. The licensee, therefore, concluded that an additional four months of operation was justified with selective plugging of tubes in the affected hard spot regions.

Also, in the February 4, 1977 submittal, the licensee provided a program of combined analytical, experimental, and field data acquisition for the long term resolution of the entire denting issue. A Westinghouse status report will be complete in March 1977, and, thereafter, further plugging or a change in the operational mode would be implemented, as required, on a case by case basis.

EVALUATION

By letter dated February 4, 1977, the licensee (VEPCO) proposed to continue power operation of Surry Unit No. 1 for an additional four (4) months beyond the twenty (20) equivalent power days approved by the NRC on January 19, 1977. This proposal was based upon the data and the supporting conclusions discussed previously. 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 within the proposed four months of operation because of the following:

a. Laboratory examinations of 71 tubes removed from Surry Units 1 and 2 and Turkey Point Unit 4 steam generators indicate that cracking was confined only to row one tubes.

b. All tubes in row one are plugged.

c. Effective U-bend strain is 30-50% lower in row 2 than in row 1.

d. Flow slot blocking devices have been inserted to arrest further flow slot closure and thus prevent further anticlastic straining of the U-bend tubes.

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 insertion of the flow slot blocking devices additional loads are 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 load 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. Since the total area of all six flow slots is only a small fraction of the total area for flow circulation, the effect of the flow slot blocking devices and hourglassing on the thermal hydraulic performance of the steam generator will be negligible. There will be a slight decrease in the circulation ratio and the liquid flow velocities, with an increase in raw steam quality. But these are so small that they may be disregarded.

4. With regard to the anti-vibration bar degradation problem revealed during the inspection of San Onofre Unit No. 1 steam generators, there is no reason to believe that similar problems will occur at Surry Unit No. 1 because there are basic differences in both the design and the material used. Anti-vibration bars in Surry Unit 1 steam generators are made of Inconel 600 instead of carbon steel, of square cross-section instead of round, and have smaller clearances than those originally employed at San Onofre Unit No. 1.

5. Since the time that Surry established AVT chemistry control, wastage and caustic stress corrosion cracking experience has been quite satisfactory. No substantial tube degradation from these corrosion mechanisms is expected to occur during normal operation.

Consideration of Continued Denting: With respect to the effect of continued support plate expansion on continued tube denting, however, the staff does not agree that the proposed additional four (4) months of power operation can be fully justified on the basis of the information thus far submitted. The licensee has been unable to quantify the effects on tubes at intersections due to the continuing growth of magnetite. Therefore, concern over a possible increase in tube failures at the tube/support intersections cannot be completely alleviated. Also, because of the absence of an explicit plugging criteria directed toward tubes subjected to increased plate strain, there is some concern that the integrity of some un-plugged dented tubes cannot be maintained during postulated accidents.

There are, however, several factors which support a shorter operating period with more stringent operating conditions while additional information is being generated; i.e., qualitative and preliminary quantitative integrity data, the low consequences of the relatively limited tube leakage that would be expected under postulated accident conditions and the very low probability of an initiating accident

coincident with large numbers of tube failures.

The qualitative and preliminary quantitative integrity data is summarized as follows:

a. Preliminary analyses of the support plate expansion (with a flow slot blocking device) indicated small hard spot strain increases and plate perimeter deformations.
b. Most of the tubes in hard spot regions and those that have leaked previously have been plugged, based on the logic derived from past experience. 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. Possible 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, until the crack grows 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.

Regarding the consequences of tube failures postulated under loadings imposed by independently initiated transients or accidents, scoping calculations which postulate such additional tube failures have been performed by the staff to obtain perspective on the magnitude of the potential hazard and to determine the degree of assurance of tube integrity required.

The two accident events which result in loadings significantly different than those seen in normal operation are the loss of coolant accident (which results in external, or collapse, forces on the steam generator tubes) and the steam line break accident (which results in internal, or burst, forces on the steam generator tubes). The potential safety significance of steam generator tube failures during a loss of coolant accident is that steam entering the primary system through failed tubes could cause a back-pressure which would retard the entry of emergency core cooling water into the core. The potential safety significance of steam generator tube failures during a steamline break accident is that radioactivity normally retained within the primary coolant system could be released to the environment. This radioactivity could include radioactivity in the primary coolant prior to the accident and radioactivity released from the fuel during the fuel temperature transient resulting from the accident.

The staff assessment of LOCAs with tube leakage shows that relatively small leakages (less than about 5 gpm) are within the typical uncertainty in computing the reflooding rate for approved ECCS performance calculations and do not warrant concern. Greater leakages (less than about 50 gpm) show a measurable effect of several percent on allowable nuclear peaking factors. However, the normal margin between actual operating peak power distribution and the allowable power peaking limits precludes exceeding the criteria of 10 CFR Part 50.46.

For the case of a postulated main steamline break, we estimate, using conservative assumptions (Table 1), that steam generator tube leak rates as high as 250 gpm may be tolerated without exceeding a 2-hour dose to the thyroid of 150 rem at the site boundary. This calculation was performed under the assumption that the coolant activity limits specified in the Standard Technical Specifications for Westinghouse plants are in force at the Surry Unit 1 plant . Such activity limits and sampling requirements have been

set forth in the Order and details are attached hereto as Appendix B.

In addition, to assure that our assumptions on the course of such a main steamline break accident are valid, the licensee has committed to review its operator procedures for this event and confirm that the operator will have the information and instructions to depressurize the primary system, thus stopping any leak through the tubes, within about 30 minutes of initiation of this sequence of events.

We have also considered the less probable event in which the steamline and steam generator tube failures occur at a time when the plant is operating with high coolant activity resulting from previous power level changes. As a result, we are imposing limitations on the operation of Surry Unit No. 1 to limit the allowable primary coolant iodine activity following power transients to 10.0μCi/g of Dose Equivalent I-131. This will provide assurance that, were the steamline break with resulting tube leaks not exceeding 250 gpm postulated to occur at such a time, the calculated doses would not exceed the 10 CFR Part 100 guidelines (Table 2).

However, for the reasons outlined elsewhere in this report, tube failures with high leak rates are not anticipated for the types of cracking associated with tube denting. Moreover, the additional limitations developed by the NRC staff assure that the development of degradation in tube integrity will be detected and operation terminated before it becomes significant. For these reasons, even under accident conditions we would not expect leakage in excess of about 50 gpm. The dose consequence discussed above would be reduced accordingly.

Additionally, we have estimated the failure probability of breaks in the primary or secondary system piping which might lead to conditions imposing severe stresses on the steam generator tubes thus potentially causing failures. In this analysis we considered the probabilities associated with large pipe breaks because we conclude that significant loads and high radiological consequences resulting from small breaks in the primary or secondary system are less likely than from large pipe breaks. Transient forces under small break conditions are smaller and larger margins exist with respect to significant consequences given an event with additional tube leakage. For example, the failure to close a single safety valve during a transient would result in a depressurization of the secondary system over many minutes (as opposed to about two minutes for a large pipe break) and would not result in uncovering the steam generator tubes (providing some washout of iodine from any primary system coolant flashed into the secondary system through tube failures).

We estimated the failure probability of large pipes in the primary and secondary system to be that associated with piping greater than six inches in diameter as given in WASH-1400. The median value given in WASH-1400 for such piping is 10^{-4} per plant per year with an uncertainty spread of from 10^{-3} to 10^{-5} per plant per year. In addition, the short period of time (60 days) that the facility will operate prior to a further determination on steam generator integrity margins will significantly reduce the likelihood of an unacceptable event. On this basis, we conclude that the likelihood of a large pipe failure leading to a severe steamline break or loss of coolant accident (LOCA) during 60 days of operation is on the order of one chance in fifty thousand (2×10^{-5}). The probability of a LOCA or a steamline break leading to large additional leakage (more than about 250 gpm) from the primary to secondary system is significantly lower than this value.

Additional 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 thru-wall cracks, the staff has developed certain additional operating limitations. These limitations are designed to assure, in the absence of adequate analytical assessment thus far, the detection of the onset of tube degradation before it becomes of imminent safety concern.

A reduction of the plugging limit for primary to secondary leakage, described below, will assure that no individual crack 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 no more than two (2) tube leaks per plant during any twenty (20) day period. This restriction, by limiting the potential number of heat up and cool down cycles resulting from tube plugging, also minimizes concern for possible thermal ratcheting.

While these conditions assure an adequate level of integrity for short term operation, more complete analyses of the structural effects of continued tube support plate corrosion is needed in order to provide adequate assurance of continued integrity for larger periods beyond an additional 60 days after February 8, 1977. The staff requirements

for additional information are described below. When such information is received it will be assessed by the staff to determine whether or not the facility can continue to operate beyond the additional 60 days considered herein without further inservice inspection.

The specific limitations developed by the staff are:

a. The leakage limits shall be reduced from 1.0 to 0.3 gpm per steam generator (See Appendix A).

b. The concentration of iodine in the primary coolant shall be limited to 1.0μCi/gram during normal operation and to 10μCi/gram Dose Equivalent I-131 during power transients (See Appendix B).

c. Reactor operation will be terminated if primary to secondary leakage which is attributable to 2 or more tubes per plant occurs during a 20 day period. Leakage means the occurrence of measurable activity on the secondary side which is identified as a leak. Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

The information requested in Appendix C will be supplied by the licensee within 45 days of the date of this order. Upon completion of the NRC staff review of this data a determination will be made whether to allow continued operation or whether to require cold shutdown to perform inservice inspection and/or to provide additional supportive data.

CONCLUSIONS

For the foregoing reasons, we have concluded that:

1. the limitations set forth herein will provide reasonable assurance that the public health and safety will not be endangered by operation of the facility in the manner described herein,
2. such activities will be conducted in compliance with the Commission's regulations, and will not be inimical to the common defense and

security or to the health and safety of the public.

Date: February 11, 1977

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TABLE 1

ASSUMPTIONS USED IN ANALYSIS OF POSTULATED MAIN STEAM LINE FAILURE WITH LARGE STEAM GENERATOR TUBE LEAKS

1. Reactor Power = 2546 Mwth

2. Steam Generator tube leak of 250 gpm

3. Leak stops after 30 minutes

4. Iodine spiking factor of 500

5. Meteorological conditions corresponding to a 30-meter elevated release with fumigation and 0.4 m/sec wind speed at a distance of 503 meters ($X/Q = 1.7 \times 10^{-3}$ sec/m^3).

6. Primary coolant activity prior to the accident of 1. μCi/g of Dose Equivalent I-131.

TABLE 2

ASSUMPTIONS USED IN ANALYSIS OF
POSTULATED MAIN STEAM LINE FAILURE
WITH LARGE STEAM GENERATOR TUBE LEAKS
AND WITH A PRIOR IODINE SPIKE

1. Reactor Power = 2546 Mwth

2. Steam Generator tube leak of 250 gpm

3. Leak stops after 30 minutes

4. Iodine spiking factor of 500

5. Meteorological conditions corresponding to a 30-meter elevated release with fumigation and 0.4 m/sec wind speed at a distance of 503 meters ($\chi/Q = 1.7 \times 10^{-3}$ sec/m^3).

6. Primary coolant activity prior to the accident of 10. μCi/g of Dose Equivalent I-131

APPENDIX A

LIMITS ON PRIMARY TO SECONDARY COOLANT LEAKAGE

1. In addition to the limits set forth in the Technical Specifications for this facility, the primary system leakage shall be limited to assure that total primary to secondary leakage through all steam generators not isolated from the Reactor Coolant System shall not exceed 1 gpm nor 0.3 gpm through any one steam generator not isolated from the Reactor Coolant System.
2. For detection of leakage exceeding the rate limits the tubes shall be plugged and the incident reported to the NRC.

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. $\leq$ $41/\bar{E}$ 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 > $41/\bar{E}$ 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 > $41/\bar{E}$ 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.

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.

APPENDIX C

REQUEST FOR INFORMATION

1. The Licensee should correlate the effects of tube support plate expansion on the strain in the tubes in the "hard spot" regions, develop a plugging criteria, before and after blocking the flow slots, and provide the basis for the plugging criteria.

2. Provide an estimate of U-bend residual stresses for tubes in rows 2, 3 and 4.

3. Provide a summary of the Westinghouse experimental programs regarding denting, intergranular stress assisted corrosion, corrosion rate, the results to date, and the schedules and milestones for future work or implementation of any further plugging.

4. Provide material and processing specifications as requested in Question 23 of January 21, 1977, letter R.W. Reid to W.L. Proffitt.

5. Provide main steam line break and loss of coolant accident consequence analysis and the justification for an assumed number of tube failures concurrent with the event and resultant leakage.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

CORRECTIVE ORDER

On February 8, 1977, the Nuclear Regulatory Commission issued an Order for Modification of License in the captioned matter. Said Order contained a typographical error in provision number 4. This Corrective Order will rectify such error.

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 revising provision 4, of Section III of the Order for Modification of License, in the captioned matter, dated February 8, 1977, to read as follows:

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 Safety Evaluation.

FOR THE NUCLEAR REGULATORY COMMISSION

Ben C. Rusche, Director
Office of Nuclear Reactor Regulation

Dated in Bethesda, Maryland
this 11 day of February 1977