November 5, 1986

Docket No. 50-395

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OGC-Bethesda	Tech Branch
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L. Tremper, LFMB	Gray File

Mr. D. A. Nauman
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
P.O. Box 764 (Mail Code 167)
Columbia, South Carolina 29218

Dear Mr. Nauman:

The Commission has issued the enclosed Amendment No. 54 to Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated January 16 1986, as revised August 15 and September 15, 1986, and supplemented May 8, and October 20, 1986.

The amendment changes the Technical Specifications to modify steam generator tube plugging requirements for tube defects located in the tubesheet region. The amendment is effective as of its date of issuance and until the end of the fifth refueling outage.

A copy of the related Safety Evaluation and Federal Register notice are enclosed.

Sincerely,

/S/

Jon B. Hopkins, Project Manager
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 54 to NPF-12
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

*See previous concurrence

*LA:PAD#2	*PM:PAD#2	*OGC	*PD:PAD#2
DMiller	JHopkins:hc		LRubenstein
10/ /86	10/ /86	10/ /86	10/ /86

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The amendment changes the Technical Specifications to modify steam generator tube plugging requirements for tube defects located in the tubesheet region. The amendment is effective as of its date of issuance and until the end of the fifth refueling outage.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

/s/

Jon B. Hopkins, Project Manager
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 54 to NPF-12
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PAD#2	PM:PAD#2	OGC w/ noted revision	PD:PAD#2
DMiller	JHopkins:hc	MYoung	LRubenstein
10/ /86	10/28/86	10/30/86	10/ /86

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

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 54
License No. NPF-12

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by South Carolina Electric & Gas Company and South Carolina Public Service Authority (the licensees) dated January 16, 1986, as revised August 15, 1986, and September 15, 1986, and supplemented May 8 and October 20, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-12 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 54 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This amendment is effective as of its date of issuance and until the end of the fifth refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

L. S. Rubenstein

Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 5, 1986

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 54 TO FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. Corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 4-12	3/4 4-12
3/4 4-14	3/4 4-14
3/4 4-15	3/4 4-15
B3/4 4-3	B3/4 4-3

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All nonplugged tubes that previously had detectable wall penetrations greater than 20%.

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.5.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. In addition to the sample required in 4.4.5.2 b.1 through 3, all tubes which have had the F* criteria applied will be inspected in the tubesheet region. These tubes may be excluded from 4.4.5.2 b.1 provided the only previous wall penetration of >20% was located below the F* distance.

d. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness. This definition does not apply to the area of the tubesheet region below the F* distance provided the tube is not degraded (i.e., no indications of cracking) within the F* distance.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

SURVEILLANCE REQUIREMENTS (Continued)

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

10. F* Distance is the distance into the tubesheet from the face of the tubesheet or the top of the last hardroll, whichever is lower (further into the tubesheet) that has been conservatively chosen to be 1.6 inches.

11. F* TUBE is the tube with degradation equal to or greater than 40%, below the F* distance and not degraded (i.e., no indications of cracking) within the F* distance. The application of F* expires at the end of the fifth fuel cycle.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to 10 CFR 50.72(b)2(i) prior to resumption of plant operation. A report pursuant to 10 CFR 50.73(a)2(ii) shall be submitted to provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. The results of inspections of F* tubes shall be reported to the Commission in a report to the Director, ONRR, prior to the restart of the unit following the inspection. This report shall include:

1. Identification of F* tubes, and
2. Location and size of the degradation

NRC approval of this report is not required prior to restart.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage-type degradation that has penetrated 20% of the original tube wall thickness.

For the tubes with degradation below the F* distance, but not degraded within the F* distance, plugging is not required.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to 10 CFR 50.72(b)2(i) prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 54 TO FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-395

INTRODUCTION

By letter dated January 16, 1986, as revised August 15, and September 15, 1986, and supplemented May 8, and October 20, 1986, South Carolina Electric and Gas Company (the licensee for V. C. Summer Nuclear Station) requested a revision to the Technical Specifications (TS), Section 3/4.4.5., "Steam Generators." This revision seeks to change the plugging limit definition in Item 4.4.5.4.a. and would exclude from plugging those tubes with indications approximately 1.6 inches or greater below the top of the tubesheet provided that the top 1.6 inches of the tube within the tubesheet is not degraded. Westinghouse Reports WCAP 11228 and WCAP 11229 "Tubesheet Region Plugging Criterion," which were part of the TS amendment request, address the issue of repairing or plugging full depth hardroll expanded steam generator tubes which may have experienced degradation within the tubesheet area and provide the technical justification for the licensee's TS change request. WCAP 11229 is a nonproprietary version of WCAP 11228.

Existing plant TS tube plugging criteria apply throughout the tube length and do not take into account the reinforcing effect of the tubesheet on the external surface of the tube. The presence of the tubesheet will constrain the tube and will complement its integrity in that region by precluding tube deformation beyond its expanded outside diameter. The resistance to both tube rupture and tube collapse is significantly strengthened by the tubesheet. In addition, the proximity of the tubesheet significantly affects the leak behavior of throughwall tube cracks in this region, i.e., no significant leakage relative to plant TS allowables is to be expected. Based on these these considerations, the use of an alternate criterion for plugging is justified.

The purpose for the development of the proposed criterion is to obviate the need to remove a tube from service (by plugging) due to detection of indications, generally by eddy current testing (ECT), in a region extending over most of the length of tubing within the tubesheet. This safety evaluation assesses the integrity of the tube bundle with ECT indications on tubes within the tubesheet under normal operating and postulated accident conditions.

The proposed criterion identifies a distance, designated P* in the original submittal, now designated as F* and referred to as the F* criterion, below the top of the tubesheet below which tube degradation of any extent does not necessitate plugging. The criterion, according to the licensee's evaluation, provides the same level of protection for tube degradation in the tubesheet region as that afforded by Regulatory Guide (RG) 1.121 for degradation located outside the tubesheet region. Limitations on the use of the criterion have also been discussed by the licensee.

The Federal Register Notice (51 FR 9907, March 21, 1986) stated that under the amendment steam generator tube imperfections would be addressed by the P-STAR evaluation method. This evaluation method relied on the combination of an adjacent tube and the tube-to-tubesheet interface to determine the distance below which tube imperfections need not be plugged. The final evaluation method approved by this amendment is called F-STAR and is essentially the same. The F-STAR method just uses the tube-to-tubesheet interface to determine the distance below which imperfections need not be plugged and does not include consideration of an adjacent tube. This results in a slightly greater distance of tube-to-tubesheet interface necessary to preclude plugging and results in fewer imperfections being addressed by the evaluation method. Therefore, this action is not being renoticed because the subject matter of the amendment does not significantly differ from that originally noticed.

EVALUATION OF TUBE PLUGGING CRITERION

Engagement Distance Determination

The licensee determined a distance below the top of the tubesheet below which tube degradation of any extent does not necessitate plugging. This criterion would be used in determining whether or not plugging of full depth hardroll expanded steam generator tubes is necessary for degradation which has been detected in that portion of the tube which is within the tubesheet.

The proposed criterion forms the basis for obviating the need to remove a tube from service (by plugging) due to detection of indications, e.g., by eddy current testing (ECT), in a region extending over most of the length of tubing within the tubesheet. This evaluation applies to the V. C. Summer Westinghouse Model D3 steam generators and assesses the integrity of the tube bundle for tube ECT indications occurring in the length of tubing within the tubesheet, relative to:

1) Maintenance of tube integrity for all loadings associated with normal plant conditions, including startup, operation in power range, hot standby and cooldown, as well as all anticipated transients.

2) Maintenance of tube integrity under postulated limiting conditions of primary-to-secondary and secondary-to-primary differential pressure, e.g., steamline break (SLB) or feedwater line break (FLB).

3) Limitation of primary-to-secondary leakage consistent with accident analysis assumptions.

The criterion provides for sufficient engagement of the tube to tubesheet hardroll such that pullout forces that could be developed during normal or accident operating conditions would be successfully resisted by the elastic preload between the tube and tubesheet even in the event of a circumferential break in the tube below the distance.

In order to evaluate the applicability of any developed criterion for indications within the tubesheet, some postulated type of degradation must necessarily be considered. For this evaluation it was postulated that a circumferential severance of a tube could occur, contrary to existing plant operating experience. However, implicit in assuming a circumferential severance to occur, is the consideration that degradation of any extent could be demonstrated to be tolerable below the location determined acceptable for the postulated condition.

When the tubes have been hardrolled into the tubesheet, any axial loads developed by pressure and/or mechanical forces acting on the tubes are resisted by frictional forces developed by the elastic preload that exists between the tube and the tubesheet. For some specific length of engagement of the hardroll, no significant axial forces will be transmitted further down the tube, and that length of tubing, i.e., F*, will be sufficient to anchor the tube in the tubesheet. In order to determine the value of F* for application in Model D3 steam generators, a testing program was conducted to measure the elastic preload of the tubes in the tubesheet.

Tubes are installed in the steam generator tubesheet by a hardrolling process which expands the tube to bring the outside surface into intimate contact with the tubesheet hole. The roll process and roll torque are specified to result in a metal-to-metal interference fit between the tube and the tubesheet.

A test program was conducted by Westinghouse to quantify the degree of interference fit between the tube and the tubesheet provided by the full depth mechanical hardrolling operation. The data generated in these tests has been analyzed to determine the length of hardroll required to preclude axial tube forces from being transmitted further along the tube, i.e., to establish the F* criterion. The amount of interference was determined by installing tube specimens in collars specifically designed to simulate the tubesheet radial stiffness. The test configuration consisted of six cylindrical collars. A mill annealed, Inconel 600 (ASME SB-163) tubing specimen was hardrolled into each collar using a process which simulated actual tube installation conditions.

Once the hardrolling was completed, the test collars were removed from the tube specimens and the springback of the tube was measured. The amount of springback was used in an analysis to determine the magnitude of the interference fit, which is, therefore, representative of the residual tube to tubesheet radial load in Westinghouse Model D steam generators.

During plant operation the amount of preload will change depending on the pressure and temperature conditions experienced by the tube. The room temperature preload stresses, i.e, radial, circumferential and axial, are such that the material is nearly in the yield state if a comparison is made to ASME

Code minimum material properties. Since the coefficient of thermal expansion of the tube is greater than that of the tubesheet, heatup of the plant will result in an increase in the preload and could result in some yielding of the tube. In addition, the yield strength of the tube material decreases with temperature. Both of these effects may result in the preload being reduced upon return to ambient temperature conditions, i.e., in the cold condition. However, based on the results obtained from the pullout tests, this is not expected to be the case as even with a very high temperature relaxation soak the results show the analysis to be conservative.

The plant operating pressure influences the preload directly based on the application of the pressure load to the inner diameter of the tube, thus increasing the amount of interface loading. The pressure also acts indirectly to decrease the amount of interface loading by causing the tubesheet to bow upward. This bow results in a dilation of the tubesheet holes, thus, reducing the amount of tube to tubesheet preload. Each of these effects was quantitatively treated.

Analytically combining the room temperature hardroll preload with the thermal, pressure, and tubesheet bow effects resulted in a net positive operating preload during normal and faulted operation. In addition to restraining the tube in the tubesheet, this preload should effectively retard leakage from indications in the tubesheet region of the tubes.

The applied loads to the tubes which could result in pullout from the tubesheet during all normal and postulated accident conditions are predominantly axial and due to the internal to external pressure differences. For a tube which has not been degraded, the axial pressure load is given by the product of the pressure with the internal cross-sectional area. However, for a tube with internal degradation, e.g., cracks oriented at an angle to the axis of the tube, the internal pressure may also act on the faces of the cracks. Thus, for a tube which is conservatively postulated to be severed at some location within the tubesheet, the total force acting to remove the tube from the tubesheet is given by the product of the pressure and the cross-sectional area of the tubesheet hole. Any other forces such as fluid drag forces in the U-bends and vertical seismic forces are negligible by comparison.

The calculation of the required engagement distance is based on determining the length for preload frictional forces to equilibrate the applied operating loads. The axial friction force was found as the product of the radial preload force and the coefficient of friction between the tube and the tubesheet. The value assumed for the coefficient of friction was for sliding of nickel on mild steel under "greasy" conditions.

For the maximum normal pressure applied load with a safety factor of 3, the length of hardroll required is exceeded by the V. C. Summer value for F* of 1.6 inches.

Similarly, the required engagement length for faulted conditions using a safety factor corresponding to a ASME Code safety factor of 1.0/0.7 for allowable stress for faulted conditions is similarly exceeded by the V. C. Summer F* value.

The F* value thus determined for the required length of hardroll engagement below the BRT or the top of the tubesheet, whichever is greater relative to the top of the tubesheet, is sufficient to resist tube pullout during both normal and postulated accident condition loadings. Furthermore, the uncertainty in position of the ECT indication must be added to the criterion for the final calculation of F*. A conservative allowance for uncertainty in ECT position indication is available in the F* distance of 1.6 inches in the V. C. Summer Technical Specifications on Steam Generators, Section 3/4.4.5.

Rolled Tube Pullout Tests

The engagement distance determination discussed above was calculated from a derived preload force and an assumed static coefficient of friction for tube to tubesheet contact. A direct measurement of this static coefficient of friction is difficult. However, a simple pull test on a rolled tube joint provided both support for the derived preload force (less the effects of thermal expansion and internal pressure tightening) and an indirect measurement of the static coefficient of friction. The results of the testing verify the calculation as being conservative.

Pullout tests were conducted on several actual rolled joints with various amounts of wall thinning. As with the preload tests, the test configuration consisted of mill annealed, Inconel 600 (ASME SB-163) Model D3 tubing, hardrolled into carbon steel collars with an outside diameter to simulate tubesheet rigidity. Inside surface roughness values of the collars were measured and recorded. The specification of surface roughness for the fabrication of the collars was the same as that used for the fabrication of the Model D tubesheets. After rolling, an inside circumferential cut was machined through the wall of the tube at a controlled distance from the bottom of the hardroll transition (opposite the tube weld). The machined cut simulated a severed tube condition. To simulate any possible effect of reduced preload force due to tube yielding during manufacturing heat treatment and during reactor operation, the samples were subjected to a heat soak.

Based on the observed pullout forces, the coefficient of friction assumed in the engagement distance determination was verified to be conservative.

Rolled Tube Hydraulic Proof Tests

Similar to the rolled tube pullout tests, pressure tests were conducted on rolled joints and with nominal degrees of wall thinning. As with the preload and pullout tests, the test configuration consisted of mill annealed, Inconel 600 (ASME SB-163) Model D3 tubing, hardrolled into carbon steel collars with an outside diameter to simulate tubesheet rigidity. As with the pullout test samples, a machined cut was used to simulate a severed tube condition. To simulate any possible effects of reduced preload force due to tube yielding during manufacturing heat treatment, these samples were also subjected to a heat soak. The pressure tests were performed at room temperature using water.

These proof tests showed that even for rolled joints considerably less than the F* distance in length at less than nominal wall thinning, pressure induced axial forces of several thousands of pounds or greater are necessary to cause the tube to release from the tubesheet. Thus, the preload based calculation of required engagement distance is indicated to be conservative.

Primary-To-Secondary Leakage Considerations

As described above to apply the F* criterion the applicable tube must have a certain minimum length of hardroll engagement below the top of the tubsheet or the BRT, whichever is greater relative to the top of the tubesheet. For V. C. Summer, South Carolina Electric and Gas has conservatively established an F* distance of 1.6 inches. The presence of the elastic preload presents a significant resistance to flow of primary-to-secondary or secondary-to-primary water for degradation which has progressed fully through the thickness of the tube wall. In effect, no leakage would be expected if a sufficient length of hardroll is present. This has been demonstrated in high pressure fossil boilers where hardrolling of tube to the tubesheet joints is the only mechanism resisting flow, and in steam generator sleeve-to-tube joints made by the Westinghouse hybrid expansion joint process. This was also confirmed by the hydraulic proof test specimens which were pressurized up to and in excess of the faulted operating conditions. Because of the difficulty in accurately sizing stress corrosion crack indications the Technical Specifications require that no indications of cracking can be present within the F* distance in tubes to which the F* criterion is applied. This requirement has the effect of preventing the start of a leak path.

Tube Integrity Under Postulated Limiting Conditions

The final aspect of the evaluation is to demonstrate tube integrity under the postulated loss of coolant accident (LOCA) condition of secondary-to-primary differential pressure. A review of tube collapse strength characteristics indicates that the constraint provided to the tube by the tubesheet gives a significant margin between tube collapse strength and the limiting secondary-to-primary differential pressure condition, even in the presence of circumferential or axial indications.

EVALUATION OF PROPOSED TECHNICAL SPECIFICATIONS

The licensee proposed Technical Specifications to implement tube plugging criterion. Based upon discussions with SCE&G personnel, revised proposed Technical Specifications (TS) were submitted in a letter dated September 15, 1986. The following addresses the changes in the TS to implement the tube plugging criterion.

1. The TS contain a definition of the F* distance, which is 1.6 inches, and a definition of a F* tube, which is a tube left in service by application of the F* criterion.

2. The TS contain a specific provision for reinspection tubesheet region of F* tubes in addition to the normal TS required sampling.

3. Special reports containing the results of inspection or reinspection of F* tubes are to be submitted to the Commission prior to restart.

4. The F* criterion or plugging limit is defined such that tubes need not be plugged because of ECT indications, equal to or greater than 40% through wall, that are below the F* distance from the top of the tubesheet (or from the top of the last hardroll whichever is lower) provided the tube is not degraded within the F* distance. The restriction on no degradation within the F* distance means that there are no indications of cracking. This restiction has been incorporated because of the difficulty in accurately sizing stress corrosion cracking. It is recognized that stress corrosion cracking that appears by ECT to be shallow may in fact be considerably deeper. In addition the engagement distance analysis and the testing program were based upon tubes that did not contain imperfections.

5. The application of the F* criterion is being approved until the end of the fifth fuel cycle. This time provision was included in the proposed TS at the request of the staff to give the staff the opportunity to review and evaluate the results of subsequent inspections before extending or revising staff approval for use of the F* criterion.

The staff has reviewed the TS changes proposed and finds them acceptable. This TS provides acceptable implementation of the tube plugging criterion as analyzed in the Westinghouse Topical Report and evaluated in this Safety Evaluation Report.

SUMMARY

Based on a review of the licensee's submittals the staff concludes that tubes can be left inservice with eddy current indications of pluggable magnitude that are below the F* distance provided the tube is not degraded within the F* distance. The F* distance is defined as 1.6 inches from the top of the tubesheet or from the top of the last hardroll whichever is lower.

From the results of the testing and analysis, it is concluded that following the installation of a tube by the standard hardrolling process, a residual radial preload stress exists due to the plastic deformation of the tube and tubesheet interface. This residual stress is expected to restrain the tube in the tubesheet while providing a leak limiting seal condition even if the tube is completely severed circumferentially at the F* distance below the top of the tubesheet.

The application of the F* criterion is being approved until the end of the fifth fuel cycle. The staff concludes that these proposed Technical Specification changes on Steam Generators, Section 3/4.4.5 are acceptable.

ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32, the Commission has determined that issuance of the amendment will have no significant impact on the environment (51 FR 26484, dated July 23, 1986).

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 5, 1986

Principal Contributors:

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U.S. NUCLEAR REGULATORY COMMISSION

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 54 to Facility Operating License No. NPF-12, issued to South Carolina Electric and Gas Company, and South Carolina Public Service Authority (the licensees), which revised the Technical Specifications for operation of the Virgil C. Summer Nuclear Station, Unit 1 (the facility) located in Fairfield County, South Carolina. The amendment is effective as of the date of issuance and until the end of the fifth refueling outage.

The amendment changes the Technical Specifications to modify steam generator tube plugging requirements for tube defects located in the tubesheet region.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Prior Hearing in connection with this action in the FEDERAL REGISTER on March 21, 1986 (51 FR 9907). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment and Finding of No Significant Impact (51 FR 26484) related to the action and has concluded that an environmental impact statement is not warranted because there will be no environmental impact attributable to the action beyond that which has been predicted and described in the Commission's Final Environmental Statement for the facility dated May 1981.

For further details with respect to the action see (1) the application for amendment dated January 16, 1986, revised August 15, and September 15, 1986, and as supplemented May 8, and October 20 1986, (2) Amendment No.54 to License No. NPF-12, and (3) the Commission's related Safety Evaluation and Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., and at the Fairfield County Library, Garden and Washington Streets, Winnsboro, South Carolina 29810. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of PWR Licensing-A.

Dated at Bethesda, Maryland this 5th day of November, 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

L S Rubenstein

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