March 30, 1987

Docket No. 50-395

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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. 60 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 June 27, 1986, as supplemented November 21, 1986, and February 25, 1987.

The amendment reduces the reactor coolant system flow measurement uncertainty from 3.5% to 2.1%. This amendment is effective as of its date of issuance, and shall be implemented within 30 days of issuance.

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. 60 to NPF-12
2. Safety Evaluation

cc w/enclosures:
See next page

Done 3/24

w/ comments to SE

Bep for

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3/17/87	3/17/87	3/18/87	3/17/87

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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. 60
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 June 27, 1986, as supplemented November 21, 1986, and February 25, 1987, 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. 60 , 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 shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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: March 30, 1987

ATTACHMENT TO LICENSE AMENDMENT

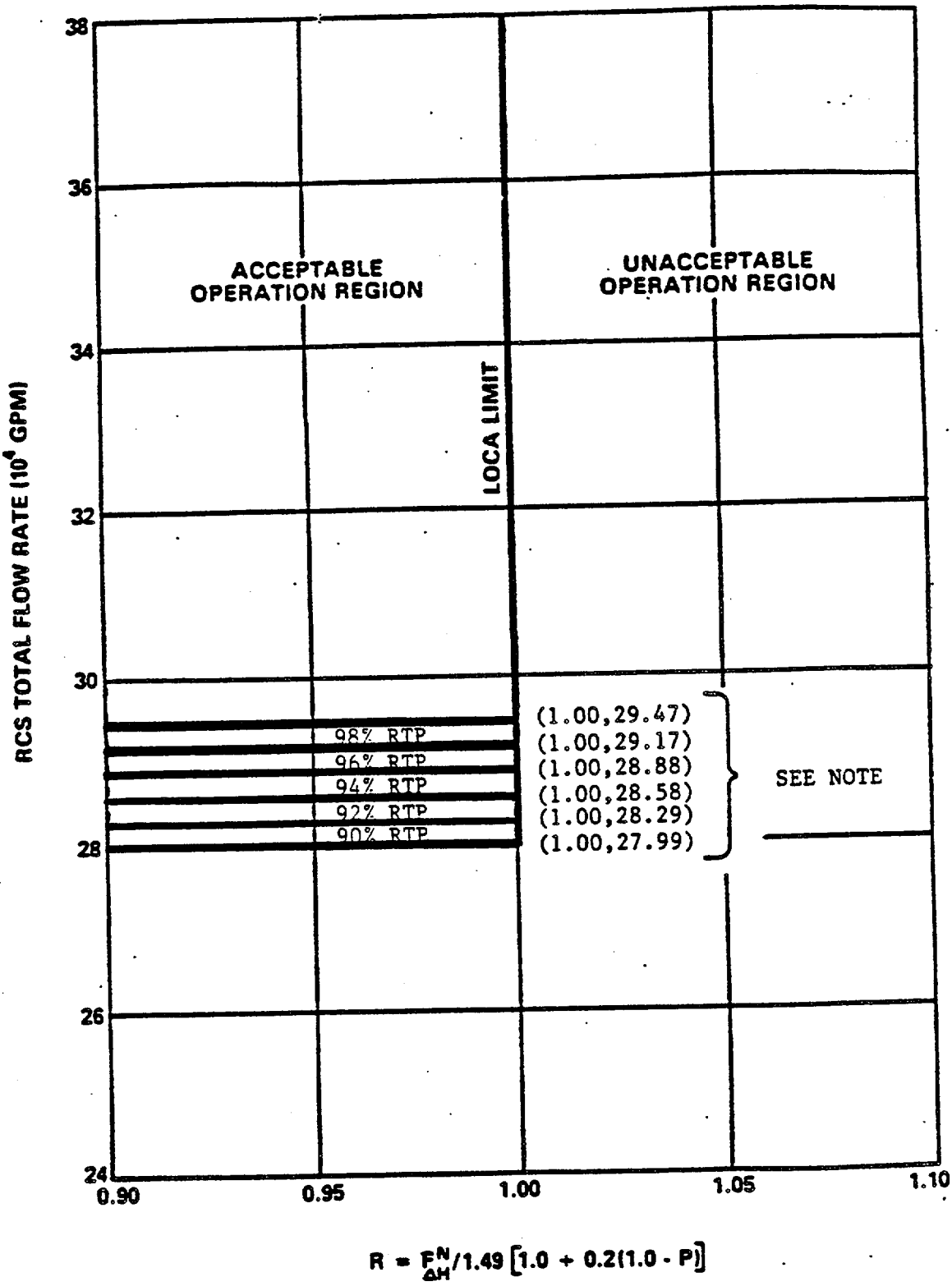
AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NO. NPF-12

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 2-8	3/4 2-8
3/4 2-10	3/4 2-10
B3/4 2-5	B3/4 2-5

MEASUREMENT UNCERTAINTIES OF 2.1% FOR FLOW AND 4.0% FOR INCORE MEASUREMENT OF $F_{\Delta H}^{N}$ ARE INCLUDED IN THIS FIGURE



$$R = F_{\Delta H}^{N}/1.49\left[1.0 + 0.2(1.0 - P)\right]$$

FIGURE 3.2-3 RCS TOTAL FLOW RATE VS. R THREE LOOP OPERATION

NOTE: When operating in this region, the restricted power levels shall be considered to be 100% of rated thermal power (RTP) for Figure 2.1-1.

POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figure 3.2-3 for 3 loop operation.

Where:

a. $R = \frac{F^N_{\Delta H}}{1.49\,[1.0 + 0.2\,(1.0 - P)]}$,

b. $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes measurement uncertainties of 2.1% for flow and 4% for incore measurement of $F^N_{\Delta H}$, and

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or
2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per specification 6.9.1.11 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

When RCS flow rate and $F_{\Delta H}^{N}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3. Measurement errors of 2.1% for RCS total flow rate and 4% for $F_{\Delta H}^{N}$ have been allowed for in determining the limits of Figure 3.2-3.

The 12 hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbles. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 60 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 June 27, 1986 (Ref. 1), South Carolina Electric and Gas Company (the licensee) requested changes in the Technical Specifications (TS) for V. C. Summer Nuclear Station to reflect the results of an RCS flow measurement uncertainty analysis. The analysis provided the basis for reducing the flow measurement uncertainty from 3.5% to 2.1% . By letters dated November 21, 1986 (Ref. 5) and February 25, 1987 (Ref. 6), the licensee supplemented the application by providing a clarification and a FSAR revision concerning the inspection of feedwater flow venturi nozzles. These supplemental letters did not change the action described in the Federal Register Notice or the initial no significant hazards consideration determination. Therefore, the application was not renoticed.

EVALUATION

The licensee provided an analysis for the RCS flow measurement uncertainty to support the requested value of 2.1%. This analysis used a statistical method similar to that in Ref. 2 which has been applied to other plants using the Westinghouse PWR design. The results indicated that the total precision calorimetric RCS flow uncertainty was ±1.790%. To establish the overall uncertainty, the effect of using three normalized elbow taps (1 per loop calibrated against the precision calorimetric) was included. This additional uncertainty for the elbow taps amounted to ±0.910% when using a process computer display and a slightly smaller value of ±0.898% when using a Digital Volt Meter (DVM) reading. By combining the elbow tap uncertainty with the precision calorimetric uncertainty, using the square root of the sum of squares method, the total RCS uncertainty in the analysis resulted in values of +2.008% and +2.003% respectively for the process computer display and DVM readings. This uncertainty value of approximately +2.00% was rounded up by the licensee by +0.1% to 2.1%.

Normalization of the elbow taps with the precision heat balance at each refueling is required to take advantage of the results of the measurement uncertainty analysis supporting the 2.1% value. A statement regarding this requirement was provided by the licensee for insertion in the FSAR (Ref. 5). In discussions

with the licensee, it was learned that each feedwater flow venturi meter is to be inspected before each refueling and is to be cleaned by a hydrolasing process which makes use of a high pressure, high temperature steam/water mixture. The licensee has stated (Ref. 6) that the FSAR will include a requirement for the inspection and hydrolasing of the venturi meters at each refueling outage. Because the licensee has committed to inspect and clean the venturi meters at each refueling outage, the 0.1% additional amount of uncertainty for venturi fouling is not applied in the analysis. We have reviewed the analysis and have found the 2.1% RCS flow uncertainty value to be used in TS 3/4.2.3 to be acceptable.

The TS change also modifies Figure 3.2-3, "RCS Total Flow Rate Vs R-Three Loop Operation," to reflect the 2.1% RCS flow uncertainty value. Previously (Ref. 3), the licensee had submitted an analysis to support a request to modify this figure to define allowable power levels for an RCS flow rate less than 100% of thermal design flow (TDF) with corrections for flow measurement uncertainty. A 2 to 1 power/flow tradeoff for RCS flow deficits of up to 5% was approved in Amendment No. 37 (Ref. 4). The maximum power level was to be reduced by 2% for each 1% reduction in flow in the range from 100% to 95% total flow. Figure 3.2-3 was modified to show the allowable RCS flow for reduced Rated Thermal Power (RTP) up to 10% (in the range from 100% to 90% RTP) in increments of 2% RTP. However, with the present request (Ref. 1) for reduction of RCS flow measurement uncertainty from 3.5% to 2.1%, the previous TS values for RCS flow for this figure are altered and need to be changed. These changes are included. It is noted that the TDF was reduced from 98,000 gpm per loop to 96,200 gpm (288,600 gpm for 3 loops) in Amendment No. 45 to the TS when the Westinghouse BART Evaluation Model was adopted. We have examined the proposed changes in Figure 3.2-3 and have found the changes to be in agreement with the effect of the new measurement uncertainty value of 2.1%. The changes are, therefore, acceptable.

TS CHANGES

The changes to the TS for V. C. Summer as a result of changing the RCS flow measurement uncertainty from 3.5% to 2.1% involve Section 3/4.2.3. These changes are discussed below:

(1) Section 3/4.2.3 - (Page 3/4 2-8)

A sentence stating that - "Figure 3.2-3 includes measurement uncertainties of 3.5% for flow" was changed to replace the 3.5% value with 2.1%. This change is acceptable for reasons explained in the evaluation.

(2) Figure 3.2-3 - RCS Total Flow Rate vs. R-Three Loop Operation - (Page 3/4 2-10)

This figure has five incremental changes in RTP from 100% down to 90% with corresponding reduction in values for RCS total flowrate in gpm down to 95%. Because of the change in RCS flow measurement uncertainty from 3.5% to 2.1%, the values shown in Figure 3.2-3 needed to be adjusted. We found these changes to be acceptable as explained in the evaluation.

(3) Bases for Section 3/4.2.3 - (Page B 3/4 2-5)

The statement in this bases - "Measurement errors of 3.5% for RCS total flow rate...." was revised to change the 3.5% to 2.1%. This agrees with the proposed change and is, therefore, acceptable. It is noted that there is no statement in the bases about the usual 0.1% penalty for venturi fouling. This has not been applied as the venturi meters are to be inspected and cleaned at each refueling. Normalization of the elbow taps with the precision heat balance at each refueling is required. The licensee has confirmed (Ref. 5 and 6) that the elbow taps will be normalized with the precision heat balance and the venturi meters will be inspected and cleaned at each refueling outage.

SUMMARY

We have reviewed the RCS flow measurement analysis performed by the licensee to justify the proposed TS changes for the V. C. Summer Nuclear Station. The RCS flow measurement analysis resulting in a measurement uncertainty of 2.1% in place of 3.5% was found to be acceptable. This change of measurement uncertainty was found to be correctly implemented in the required modifications to Figure 3.2-3 to show the 2 to 1 power/flow tradeoff. Changes were also required on pages 3/4 2-8 and B 3/4.2.3 to reflect the new flow measurement uncertainty value of 2.1%. We find that the proposed TS changes are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

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.

References

1. Letter from D. A. Nauman, South Carolina Electric and Gas Company, to H. R. Denton, NRC, dated June 27, 1986.

2. Letter from E. P. Rahe, Westinghouse Electric Corporation to C. H. Berlinger, NRC, dated March 31, 1982.

3. Letter from O. W. Dixon, Jr., South Carolina Electric and Gas Company, to H. R. Denton, NRC, dated June 19, 1984.

4. Letter from E. Adensam, NRC, to O.W. Dixon, Jr. South Carolina Electric and Gas Company, dated January 31, 1985.

5. Letter from D. A. Nauman, South Carolina Electric and Gas Company, to H. R. Denton, NRC, dated November 21, 1986.

6. Letter from D. A. Nauman, South Carolina Electric and Gas Company, to H. R. Denton, NRC, dated February 25, 1987.

Dated: March 30, 1987

Principal Contributors:

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