January 10, 1991

Docket No. 50-395

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Mr. John L. Skolds
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
P.O. Box 88
Jenkinsville, South Carolina 29065

Dear Mr. Skolds:

SUBJECT: ISSUANCE OF AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. NPF-12 REGARDING TECHNICAL SPECIFICATIONS 3/4.4.2 VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1, (TAC NO. 76867)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 94 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 May 16, 1990, as supplemented August 13, 1990.

The amendment changes the Technical Specifications by allowing a ± 3% tolerance in the lift setting of the pressurizer code safety valves (Technical Specifications 3.4.2.1 and 3.4.2.2) and by allowing heatup to Mode 3 with these valves set under cold conditions (Technical Specification 3.4.2.2 and Surveillance Requirement 4.4.2.1).

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

Sincerely,

Ronnie Lo/for

George F. Wunder, Project Manager
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 94 to NPF-12
2. Safety Evaluation

cc w/enclosures:
See next page

See Previous Concurrences

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	OGC*	:D:PD21:DRPR	:	:
NAME	:PAnderson	:GWunder:sw	: SUttal	:EAdensam	:	:
DATE	: 1 /10/91	: 1 /10/91	: / /91	: 1 /10/91	:	:

OFFICIAL RECORD COPY
Summer Amendment 76867

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. 94
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 (the licensee), dated May 16, 1990, as supplemented August 13, 1990, 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) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 94 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

Ronnie Lo/for

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 10, 1991

OFC	LA:PD21:DRPR	PM:PD21:DRPR	OGC	D:PD21:DRPR		
NAME	PAnderson:	GWunder:sw	S. Uttal	EAdensam		
DATE	~~11/28/90~~ 12/10/90	11/26/90	1/2/91	1/10/91		

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ATTACHMENT TO LICENSE AMENDMENT NO. 94

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 indicated by marginal lines.

Remove Pages	Insert Pages
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8
B3/4 4-2	B3/4 4-2

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± 3%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE residual heat removal loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The Surveillance Requirements of Specification 4.0.5 shall be met, or; the pressurizer code safety valve shall have its lift set pressure verified under cold conditions.

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 3%.*

APPLICABILITY: MODES 1, 2 and 3.#

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

#Mode 3 applicability is exempted under the following conditions:

1. There has been a least 5 days of operation in MODES 5 or 6 since the reactor was last critical, and

2. All RCCAs are fully inserted with all CRDMs de-energized.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES (PORV's)

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 94 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

1.0 INTRODUCTION

By letter dated May 16, 1990, as supplemented August 13, 1990, South Carolina Electric & Gas Company (SCE&G or the licensee), the licensee for the Virgil C. Summer Nuclear Station, Unit No. 1, (Summer Station) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. NPF-12.

The proposed amendment would (1) increase the tolerance of the pressurizer safety valve (PSV) setpoint from ±1% to ±3% and (2) allow the Mode 3 operation with one or more PSVs inoperable so that the plant can be heated up to Mode 3 condition for the purposes of testing these valves. Specifically, the request would (1) modify Limiting Conditions for Operation (LCO) 3.4.2.1 and 3.4.2.2 by changing the PSV setpoint tolerance from ±1 to ±3%, (2) modify Surveillance Requirement 4.4.2.1 to indicate that the PSV shall have its lift set pressure verified under cold conditions, and (3) add a footnote under LCO 3.4.2.2 indicating that Mode 3 applicability is exempted if the following conditions are met: (i) there have been at least 5 days of operation in Mode 5 or 6 since the reactor was last critical, and (ii) all rod cluster control assemblies (RCCA) are fully inserted with all control rod drive mechanisms (CRDM) deenergized.

2.0 EVALUATION

The Summer Station PSVs were designed and manufactured to meet the 1971 Edition including the Winter 1972 Addenda of the ASME Code, Section III, which required the PSVs to be designed to open within ± 1% of the set pressure. The current TS also impose a tolerance of ± 1% on the set pressure in the LCO for the PSVs. However, the Surveillance Requirements of these TS require testing the PSVs under Section XI of the ASME Code. 10 CFR Part 50 requires that Section XI testing be in compliance with the 1977 Edition, including the Summer 1978 Addenda of the ASME Code. This Edition of

Section XI does not specify a tolerance to be applied to lift pressure verification; therefore, the tolerance prescribed in the LCO (±1%) is used as the acceptance criteria for Section XI testing. Section XI also requires that when any valve in a system fails the setpoint criteria, additional valves in the system shall be tested, and a valve failing to function during a test shall be repaired or replaced.

The 1989 Edition of the ASME Code, Section XI, requires that the PSVs be tested per the standard ASME/ANSI OM-1987, Part 1. This standard allows the tested lift pressure to exceed the stamped set pressure by up to 3% before declaring a test failure. It also provides a guideline for testing additional valves when a valve exceeds the ±3% tolerance. Therefore, increasing the PSV setpoint tolerance to ±3% for testing acceptance criteria is in compliance with the later Code requirements.

To support the proposed TS amendments for the increased PSV setpoint tolerance and testing in Mode 3, the licensee provided the sensitivity analyses and evaluation of the existing analyses of all the transients and accidents in the Reload Transition Safety Report (RTSR) performed to determine the impacts on each transient or accident.

In evaluating the impact of the increased setpoint tolerance on the pressurization events, the PSVs were assumed to have a setpoint at the maximum tolerance of 3% plus 3% accumulation, i.e., the PSVs open at the setpoint of 2575 psia and attain the full open relief capacity at 2653 psia. The results of the evaluation indicated that (1) the low probability event of a rupture of CRDM housing, which results in the ejection of a RCCA, has the peak reactor coolant system (RCS) pressure of 2900 psia, which is below the pressure limit which would cause stresses to exceed the "Service Limit C" (an emergency condition) as defined in the ASME Code and accepted by the Standard Review Plan (SRP), Section 5.2, and (2) all other events including turbine trip and a Condition IV locked rotor event have the peak RCS pressure of 110% of the design pressure. Though the staff does not agree with the licensee's acceptance criteria of (1) 120% of the RCS design pressure for Condition IV events, and (2) the faulted condition stress limits for a rod ejection event, it finds the licensee evaluation results to be acceptable because the increased PSV setpoint tolerance limit of 3% does not result in the peak RCS pressure exceeding the SRP acceptance limits for the transients and accidents of the RTSR.

The impact of the PSV setpoint at the lower end of 3% tolerance limit was also evaluated. With a -3% tolerance, the lowest setpoint of 2425 psia remains higher than the setpoint of 2350 psia of the power operated relief valves (PORV). For the events where the departure from nucleate boiling (DNB) is of the primary concern, the analyses conservatively assumed the operation of the PORVs and the lowest set PSV since lower RCS pressure is detrimental to DNB. However, since the minimum PSV setpoint is still higher than the PORV setpoint, there is no impact on the analysis results of minimum DNB ratios.

Based on the evaluations and analyses performed, the licensee concluded that operation with PSV setpoints within ±3% tolerance about the nominal values will have no adverse impact upon the licensing basis analyses. All licensing basis criteria continue to be met and the conclusions in the RTSR remain valid. In addition, the probability of premature lifting of PSVs is not increased because of the lower PORV setpoint.

The licensee also proposed to set the PSVs under cold conditions, then heat up to Mode 3 and perform the PSV testing during Mode 3 operation using the Crosby Gage & Valve Set Point Verification Device (SPVD). This would render the PSVs inoperable during testing. The licensee, therefore, performed an examination of the impact of the Mode 3 PSV testing on all the transients and accidents assuming all PSVs inoperable. The PSV testing in Mode 3 will be allowed only after (1) at least 5 days of operation in cold shutdown (Mode 5) or a lower mode, and (2) all the RCCAs inserted with CRDMs deenergized. In addition, LCO 3.4.5 requires that the pressurizer be operable with a water volume of less than or equal to 1288 cubic feet during the operation of Modes 1, 2 and 3. This requirement ensures the presence of a steam bubble in the pressurizer. Therefore, at the time of the PSV testing, the decay heat level would be very low, no reactivity may be added to the primary side through rod motion, and there is sufficient bubble space to accommodate the reactor coolant insurge into the pressurizer. The licensee examined all the transients and accidents of the RTSR, and concluded that there was no adverse effect of the inoperable PSVs on the previously analyzed results. Therefore, the licensee's proposal to set the PSVs in the cold condition and then test them in Mode 3 is acceptable. In addition, they indicated that the use of the SPVD does not restrict the vertical movement of the spindle before, during or after testing, and that since the internal mechanism of the SPVD triggers a solenoid and releases the spindle allowing the valve to reseat, it is highly unlikely that the valve with the SPVD installed will fail in an open position, thus initiating a transient. The staff agrees with the licensee's assessment. However, as recommended by the evaluation provided with the licensee's submittal, the licensee should verify that the test procedures will assure that the probability of initiating a transient is not increased.

The staff agrees with the licensee's assessment of the proposed setpoint tolerance criteria. The staff finds that the licensee's proposed tolerance of ±3% of nominal setpoint is consistent with current versions of ASME Code requirements and is acceptable for the purpose of determining the as-found setpoint acceptability and the necessity for testing additional valves. However, because the licensee proposed to leave the setpoint of the PSVs in the ±3% range, the staff was concerned that the valves would drift beyond the ±3% value when returned to service. In response to this concern, the licensee has evaluated an additional ±3% of setpoint drift beyond the ±3% range and has determined that the limits of the accident analyses are not exceeded (Reference 3). This is a reasonable amount of additional drift which may be expected; therefore, the staff agrees that the PSVs may be left within the ±3% range following testing without resetting the valves.

The staff has reviewed the licensee's evaluation on the impacts of the proposed TS changes to allow an increased PSV set point tolerance to ±3% and to perform the PSV testing during Mode 3 operation with the conditions stated. We find that these proposed changes do not have adverse impact on the existing RTSR safety analysis results and are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment changes requirements in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the surveillance requirements. 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 off site, 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.

4.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration, which was published in the FEDERAL REGISTER on (55 FR 40474) on October 3, 1990, and consulted with the State of South Carolina. No public comments or requests for hearing were received, and the State of South Carolina did not have comments.

The staff has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Letter from O. S. Bradham (SCE&G) to USNRC, "Technical Specifications Change Request - Pressurizer Safety Valve Setpoint and Mode 3 Exception," May 16, 1990.

2. Letter from O. S. Bradham (SCE&G) to USNRC, "Modification to Technical Specification Change Request - Pressurizer Safety Valve," August 13, 1990.

3. South Carolina Electric & Gas Company Technical Work Record, "PSV Set Point Tolerance Increase," Serial No. 239-02-7834, October 15, 1990.

Dated: January 10, 1991

Principal Contributors: Y. Hsii
C. Hammer