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SCE&G®
A SCANA COMPANY

June 19, 2001
RC-01-0120

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: K. R. Cotton

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
TECHNICAL SPECIFICATION CHANGE REQUEST TSP 99-0135
TURBINE CYCLE: SAFETY VALVES

References:
1. NRC Information Notice 94-60, *Potential Overpressurization of Main Steam System*, August 22, 1994
2. Westinghouse Nuclear Safety Advisory Letter NSAL-94-001, *Operation at Reduced Power Levels with Inoperable MSSVs, January 20, 1994*
3. NUREG-1431, *Standard Technical Specifications Westinghouse Plants*, Revision 1, April 1995

South Carolina Electric and Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby requests an amendment to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) in accordance with 10CFR50.90. This proposed amendment will revise Table 3.7-1 to identify maximum allowable power range neutron flux high setpoints based on revised calculations performed in accordance with Reference 1, Reference 2, and plant specific analyses.

A change to the Bases section is submitted to reflect the appropriate calculational method.

SCE&G requests NRC review and approval of this change to the VCSNS TS by February 15, 2002, in order to implement prior to refueling outage (RF 13) currently scheduled for the Spring of 2002.

A001

The amendment request is contained in the following documents:

Attachment I	Explanation of Changes Summary Marked-up Technical Specification Pages Revised Technical Specification Pages
Attachment II	Safety Evaluation
Attachment III	No Significant Hazards Determination
Attachment IV	Environmental Impact Determination

This proposed TS amendment request has been reviewed by both the Plant Safety Review Committee and the Nuclear Safety Review Committee.

There are no other TS changes in process that will affect or be affected by this change request.

There are no significant changes to any FSAR or FPER sections. FSAR Sections 10.3 and 15.2 were reviewed. The FPER was reviewed but was not applicable.

A copy of this application and associated attachments is being provided to the designated South Carolina State official in accordance with 10 CFR 50.91.

I certify under penalty of perjury that the foregoing is true and correct.

Should you have questions, please call Mr. Jim Turkett at (803) 345-4047 or Mr. Lou Cartin at (803) 345-4728.

Very truly yours,

Stephen A. Byrne

JT/SAB/dr
Attachment(s): 4

c: N. O. Lorick
N. S. Carns
T. G. Eppink (w/o Attachment)
R. J. White
L. A. Reyes
W. R. Higgins
NRC Resident Inspector
P. Ledbetter
K. M Sutton
T. P. O'Kelley
RTS (TSP 99-0135)
File (813.20)
DMS (RC-01-0120)

STATE OF SOUTH CAROLINA :
: **TO WIT** :
COUNTY OF FAIRFIELD :

I hereby certify that on the 19th day of JUNE 2001, before me, the subscriber, a Notary Public of the State of South Carolina personally appeared Stephen A. Byrne, being duly sworn, and states that he is Senior Vice President, Nuclear Operations of the South Carolina Electric & Gas Company, a corporation of the State of South Carolina, that he provides the foregoing response for the purposes therein set forth, that the statements made are true and correct to the best of his knowledge, information, and belief, and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal James W. Turkett, Jr
Notary Public

My Commission Expires 2 OCT 2010
Date

Attachment To License Amendment No. XXX
To Facility Operating License No. NPF-12
Docket No. 50-395

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3/4 7-2	3/4 7-2
B 3/4 7-1	B 3/4 7-1

SCE&G -- EXPLANATION OF CHANGES

Page	Affected Section	Bar #	Description of Change	Reason for Change
3/4 7-2	Table 3.7-1	1	Revise power levels in second column.	To reflect maximum allowable power range neutron high flux setpoints using a combination of plant specific analyses and guidelines in NSAL 94-01
B 3/4 7-1	B 3/4.7.1.1	1	Revise description of MSSV steam relief capacity.	To properly reflect BASES for the design of the MSSV at RTP.
	B 3/4.7.1.1	2	Revise formula and setpoint derivation discussion.	To reflect maximum allowable power range neutron high flux setpoints calculated with NSAL 94-01 guidelines.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	87
2	65
3	43

INSERT NEW TABLE
INSERT "A"

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

S/G A	S/G B	S/G C		Lift Setting*		Orifice Size
XVS-2806A	XVS-2806F	XVS-2806K		1176 psig ±1%		4.515 In dia/16 sq in
XVS-2806B	XVS-2806G	XVS-2806L		1190 psig ±3%		4.515 In dia/16 sq in
XVS-2806C	XVS-2806H	XVS-2806M		1205 psig ±3%		4.515 In dia/16 sq in
XVS-2806D	XVS-2806I	XVS-2806N		1220 psig ±3%		4.515 In dia/16 sq in
XVS-2806E	XVS-2806J	XVS-2806P		1235 psig ±3%		4.515 In dia/16 sq in

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

Insert "A"

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Each Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1[1]	58[1]
2	41
3	24

Notes: 1. With one inoperable safety valve in only one operating steam generator, the Maximum Allowable Power Range Neutron Flux High Setpoint may be increased to 81% of RATED THERMAL POWER provided the predicted Moderator Temperature Coefficient is negative (< 0 pcm/°F) at hot zero power assuming all rods out and no xenon.

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 13.76 x 10^6 lbs/hr which is 110 percent of the total secondary steam flow of 12.2 x 10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 3 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 109$$

Where:

SP = Reduced reactor trip setpoint in percent of RATED THERMAL POWER.

V = Maximum number of inoperable safety valves per steam line.

109 = Power Range Neutron Flux-High Trip Setpoint for 3 loop operation.

X = Total relieving capacity of all safety valves per steam line in lbs/hour.

Y = Maximum relieving capacity of any one safety valve in lbs/hour.

INSERT "B"

Insert "B"

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is sufficient to limit secondary side pressure to within 110% of design at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are based on the plant safety analysis or are conservatively derived on the following bases, implementing Westinghouse NSAL 94-01 Methodology:

$$SP = Hi\,\phi - \xi = \left(\frac{100\%}{Q}\right)\frac{w_s \cdot h_{fg} \cdot N}{K} - \xi$$

Where:

SP = Reduced Reactor Trip Setpoint [% RATED THERMAL POWER (RTP)]

Hi ϕ = Safety Analysis Power Range High Neutron Flux Trip Setpoint (% RTP)

ξ = Power Range Neutron Flux Channel Uncertainty (% RTP)

Q = RTP plus Reactor Coolant Pump Heat (Mwt)

W_s = Minimum total steam flow rate capability of the operable main steam line code safety valves on any one steam generator at the highest MSSV opening pressure including allowances for setpoint tolerance and accumulation (lb/sec)

h_{fg} = Heat of vaporization for steam at the highest main steam line code safety valve operating pressure including allowances for setpoint tolerance and accumulation (Btu/lb)

N = Number of Loops in Plant

K = Conversion factor, 947.82 $\frac{\text{(Btu/sec)}}{\text{Mwt}}$

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Each Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1(1)	58(1)
2	41
3	24

Notes: 1. With one inoperable safety valve in only one operating steam generator, the Maximum Allowable Power Range Neutron Flux High Setpoint may be increased to 81% of RATED THERMAL POWER provided the predicted Moderator Temperature Coefficient is negative (< 0 pcm/°F) at hot zero power assuming all rods out and no xenon.

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

S/G A	S/G B	S/G C		Lift Setting *		Orifice Size
XVS-2806A	XVS-2806F	XVS-2806K		1176 psig ± 1%		4.515 In dia/16 sq in
XVS-2806B	XVS-2806G	XVS-2806L		1190 psig ± 3%		4.515 In dia/16 sq in
XVS-2806C	XVS-2806H	XVS-2806M		1205 psig ± 3%		4.515 In dia/16 sq in
XVS-2806D	XVS-2806I	XVS-2806N		1220 psig ± 3%		4.515 In dia/16 sq in
XVS-2806E	XVS-2806J	XVS-2806P		1235 psig ± 3%		4.515 In dia/16 sq in

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

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is sufficient to limit secondary side pressure to within 110% of design at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are based on the plant safety analysis or are conservatively derived on the following bases, implementing Westinghouse NSAL 94-01 Methodology:

$$SP = Hi\ \phi - \xi = \left(\frac{100\%}{Q}\right)\frac{w_s \cdot h_{fg} \cdot N}{K} - \xi$$

Where:

SP = Reduced Reactor Trip Setpoint [% RATED THERMAL POWER (RTP)]

Hi ϕ = Safety Analysis Power Range High Neutron Flux Trip Setpoint (% RTP)

ξ = Power Range Neutron Flux Channel Uncertainty (% RTP)

Q = RTP plus Reactor Coolant Pump Heat (Mwt)

W_s = Minimum total steam flow rate capability of the operable main steam line code safety valves on any one steam generator at the highest MSSV opening pressure including allowances for setpoint tolerance and accumulation (lb/sec)

h_{fg} = Heat of vaporization for steam at the highest main steam line code safety valve operating pressure including allowances for setpoint tolerance and accumulation (Btu/lb)

N = Number of Loops in Plant

K = Conversion factor, 947.82 $\frac{\text{(Btu/sec)}}{\text{Mwt}}$

SAFETY EVALUATION FOR REVISING TABLE 3.7-1 OF THE VIRGIL C. SUMMER NUCLEAR STATION TECHNICAL SPECIFICATIONS

Description of Amendment Request

South Carolina Electric & Gas Company (SCE&G) proposes a change to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) Surveillance Requirements to revise Table 3.7-1. This change will identify maximum allowable power range neutron flux high setpoints based on the plant safety analysis or conservatively derived values calculated in accordance with NRC Information Notice 94-60 and Westinghouse Nuclear Safety Advisory Letter NSAL-94-001.

Safety Evaluation

References:

1. NRC Information Notice 94-60, *Potential Overpressurization of Main Steam System*, August 22, 1994
2. Westinghouse Nuclear Safety Advisory Letter NSAL-94-001, *Operation at Reduced Power Levels with Inoperable MSSVs, January 20, 1994*

The operability of the main steam line code safety valves ensure that the secondary system pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER (RTP) coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements of the Technical Specifications based on the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The current setpoints are based on the following formula which assumes that the allowable initial power level is a linear function of the available MSSV relief capacity:

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP	=	Reduced reactor trip setpoint in % of RTP.
V	=	Maximum number of inoperable safety valves per steam line.
109	=	Power Range Neutron Flux-High Trip Setpoint for 3 loop operation.
X	=	Total relieving capacity of all safety valves per steam line in lbs/hr.
Y	=	Maximum relieving capacity of any one safety valve in lbs/hr.

As outlined in References 1 and 2, setpoints derived from the above function were determined to provide inadequate secondary side overpressure protection under certain accident conditions (i.e., a Loss of Load/Turbine Trip transient from part-power, beginning of cycle conditions). As a result, more restrictive reactor trip setpoints were defined in accordance with the recommendations of Reference 2 and imposed administratively to ensure adequate overpressure protection. This proposed change formally incorporates the more conservative calculation methods of Reference 2 into the Technical Specifications.

As recommended by Reference 2, Technical Specification 3.7.1.1 and its associated basis is modified such that the maximum power level for operation with inoperable main steam safety valves is below the heat removing capability of the operable main steam safety valves. This is conservatively imposed by setting the power range high neutron flux setpoint to this power level, thus ensuring that the actual power level cannot exceed this value. To calculate this setpoint, the governing equation is the relationship $q = m \Delta h$, where q is the heat input from the primary side, m is the steam flow rate and Δh is the heat of vaporization at the steam relief pressure. Taking into account, instrument and channel uncertainties of 9%, the following algorithm is used in defining the revised Technical Specification table setpoint values:

$$SP = Hi\ \phi - \xi = \left(\frac{100\%}{Q}\right)\frac{w_s \cdot h_{fg} \cdot N}{K} - \xi$$

Where:

SP = Reduced Reactor Trip Setpoint [% RTP]

Hi ϕ = Safety Analysis Power Range High Neutron Flux Trip Setpoint (% RTP)

ξ = Power Range Neutron Flux Channel Uncertainty (9 % RTP)

Q = RTP plus Reactor Coolant Pump Heat (Mwt)

W_s = Minimum total steam flow rate capability of the operable main steam line code safety valves on any one steam generator at the highest MSSV opening pressure including allowances for setpoint tolerance and accumulation (lb/sec)

h_{fg} = Heat of vaporization for steam at the highest main steam line code safety valve operating pressure including allowances for setpoint tolerance and accumulation (Btu/lb)

N = Number of Loops in Plant

K = Conversion factor, 947.82 $\frac{\text{(Btu/sec)}}{\text{Mwt}}$

The proposed revision to Technical Specification Table 3.7-1 includes a footnote which addresses the special case of one inoperable, secondary side safety valve in only one steam line. For this condition, it is proposed that power operation with a high flux trip setpoint of 81% be allowed provided the core's moderator temperature coefficient is negative (i.e., < 0 pcm/°F). This provision is based on plant specific analyses of the LOL/TT event performed by the Westinghouse Electric Company. Using the LOFTRAN code, a series of LOL/TT analyses were performed to examine the impact of initial power level and moderator temperature coefficient on event consequences assuming the main steam safety valve with the lowest set pressure was inoperable. It was determined that, if a moderator temperature coefficient of ≤ 0 pcm/°F is assumed, all acceptance criteria are satisfied for initial core power levels at or below 92% of RTP. Assuming a 2 % power uncertainty and 9 % uncertainty for the high flux trip channel, the proposed value for the Technical Specification Table footnote is specified at 81%.

Overall, the proposed revisions to the Technical Specification provide reduced Power Range Neutron Flux High Trip Setpoints for use during Modes 1, 2 and 3 with inoperable main steam safety valves. When implemented, the Power Range High Flux trip will restrict core power sufficiently to ensure secondary side overpressure protection during the most severe anticipated system operational transients. By using the Power Range High Flux trip to restrict core power, the Technical Specification provides an inherent margin of safety since the maximum plant operating power level would then be lower than the reactor protection system setpoint.

NO SIGNIFICANT HAZARDS DETERMINATION FOR REVISING TABLE 3.7-1 OF THE VIRGIL C. SUMMER NUCLEAR STATION TECHNICAL SPECIFICATIONS

Description of Amendment Request

South Carolina Electric & Gas Company (SCE&G) proposes a change to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) Surveillance Requirements to revise Table 3.7-1. This change will identify maximum allowable power range neutron flux high setpoints based on the plant safety analysis or conservatively derived values calculated in accordance with NRC Information Notice 94-60 and Westinghouse Nuclear Safety Advisory Letter NSAL-94-001.

Basis for No Significant Hazards Consideration Determination

South Carolina Electric & Gas Company (SCE&G) has evaluated the proposed changes to the VCSNS TS described above against the significant Hazards Criteria of 10CFR50.92 and has determined that the changes do not involve any significant hazard. The following is provided in support of this conclusion.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed changes to Technical Specification 3.7.1 and its associated bases do not contribute to the initiation of any accident previously evaluated. Supporting factors are as follows:

   - All NSSS components are compatible with the revised core power limits and resulting operating conditions. Their structural integrity is maintained during all proposed plant conditions through compliance with the ASME code.
   - Other systems important to safety are not adversely impacted and will continue to perform their design functions.
   - The revised core power limits and resulting operating conditions remain within the design envelope of the plant.

   Therefore, since the reactor coolant pressure boundary integrity and system functions are not adversely impacted, the probability of occurrence of an accident previously evaluated will be no greater than the existing design basis of the plant.

The revised method to derive allowable power levels with inoperable main steam safety valves results in lower High Flux Trip Setpoints. When implemented, the revised trip setpoints ensure that secondary system pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transients. Since the ASME and regulatory limits on secondary side overpressurization will be met, the proposed changes will not create the potential for an increase in offsite releases or doses for any accident. Therefore, there is no increase in the consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes to Technical Specification 3.7.1 and its associated bases do not introduce any new accident initiator mechanisms. Structural integrity of the RCS and the secondary side is maintained during the allowed operating conditions, and ASME code limits continue to be met during all anticipated operating conditions. In addition, no new failure modes or limiting single failure or new design requirements for auxiliary systems are being introduced. Since the safety and design requirements continue to be met and the integrity of the primary and secondary pressure boundary is maintained, no new accident scenarios have been created. Therefore, the types of accidents previously defined continue to represent the credible spectrum of events to be analyzed. A new or different kind of accident is thus not created.

3. Does this change involve a significant reduction in margin of safety?

   The proposed changes to Technical Specification 3.7.1 and its associated bases preserve the results and conclusions of plants safety analyses presented in the FSAR. The proposed changes address an identified deficiency with the current Technical Specification and, when implemented, restores the margin of safety intended. Specifically, the proposed changes ensures overpressure ensure that the secondary system pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. Therefore, there is no reduction in a margin of safety.

Pursuant to 10 CFR 50.91, the preceding analyses provides a determination that the proposed Technical Specifications change poses no significant hazard as delineated by 10 CFR 50.92.

ENVIRONMENTAL IMPACT DETERMINATION FOR REVISING Table 3.7-1 OF THE VIRGIL C. SUMMER NUCLEAR STATION TECHNICAL SPECIFICATIONS

Environmental Assessment

This proposed Technical Specification change has been evaluated against criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed change meets the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). The following is a discussion of how the proposed Technical Specification change meets the criteria for categorical exclusion.

10 CRF 51.22(c)(9): The proposed change involves a revision to Table 3.7-1. This change will identify maximum allowable power range neutron flux high setpoints based on the plant safety analysis or conservatively derived values calculated in accordance with NRC Information Notice 94-60 and Westinghouse Nuclear Safety Advisory Letter NSAL-94-001.

(I) the proposed change involves No Significance Hazards Consideration (refer to No Significance Hazards Evaluation);

(ii) there are no significant changes in the types or significant increase in the amounts of any effluents that may be released offsite since the proposed change does not affect the generation of any radioactive effluents nor does it affect any of the permitted release paths; and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Based on the aforementioned information and pursuant to 10 CFR 51.22(b), no environmental assessment or environmental impact statement need be prepared in connection with issuance of an amendment to the Technical Specifications incorporating the proposed change.