**Stephen A. Byrne**
*Senior Vice President, Nuclear Operations*
803.345.4622

SCE&G®
A SCANA COMPANY

July 23, 2003
RC-03-0155

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

ATTN: Ms. K. R. Cotton

Dear Sir/Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
LICENSE AMENDMENT REQUEST - LAR 02-3626
REACTIVITY CONTROL SYSTEMS - MODERATOR TEMPERATURE COEFFICIENT

References:

1. Robert C. Jones (NRC) letter to Nicholas J. Liparulo (Westinghouse), Acceptance for Referencing of Licensing Topical Report WCAP-13749-P, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL [End-of-Life] Moderator Temperature Coefficient Measurement," October 9, 1996

2. N. J. Liparulo (Westinghouse) letter to J. E. Lyons (NRC), Clarification of Individual Control Rod Bank Worth Benchmark Criteria in WCAP-13749, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 18, 1997

Pursuant to 10 CFR 50.90, South Carolina Electric & 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).

The proposed change will revise the near-end of life (EOL) Moderator Temperature Coefficient (MTC) Surveillance Requirement 4.1.1.3.b by placing a set of conditions on core operation, which if met, would allow exemption from the required MTC measurement. The conditional exemption will be determined on a cycle-specific basis by considering the margin predicted to the surveillance requirement MTC limit and the performance of other core parameters, such as beginning of life (BOL) MTC measurements and the critical boron concentration as a function of cycle length. The conditional exemption will improve plant availability and minimize disruptions to normal plant operations. Plant safety criteria will not be compromised by the conditional exemption of this one measurement.

This method has been accepted by the NRC (Reference 1) and clarified by Westinghouse through Amendment to the WCAP (Reference 2). The NRC has previously approved a similar TS Amendment (TAC NOs. MB5160 and MB5161) for another utility on November 26, 2002.

A001

Pursuant to 10 CFR 50.91, the enclosed analysis provides a determination that the proposed Technical Specification change poses no significant hazard as delineated by 10 CFR 50.92.

Information contained herein provides the No Significant Hazards Determination and Attachment 1 provides the TS page marked up with the proposed change. Attachment 2 provides the retyped TS page. There are no changes proposed to the Bases for TS 3/4.1.1.

No other TS revision requests are in progress which affect these pages. There are no significant changes to any FSAR or FPER sections.

The VCSNS Plant Safety Review Committee and the Nuclear Safety Review Committee have reviewed and approved the proposed change. SCE&G has notified the State of South Carolina in accordance with 10CFR50.91(b).

SCE&G requests approval of the proposed amendment by November 1, 2003, to support the VCSNS plant surveillance program. Once approved, the amendment shall be implemented within 30 days.

If you have any questions or require additional information, please contact Mr. Ronald B. Clary at (803)-345-4757.

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

7/23/03
Executed on

Stephen A. Byrne

JT/SAB/dr

Enclosures:
Evaluation of the proposed change
Attachment(s): 4
1. Proposed Technical Specification Change - Mark-up
2. Proposed Technical Specification Change - Retyped
3. List of Regulatory Commitments
4. Typical COLR Change - Mark-up (For Information Only)

c: N. O. Lorick
N. S. Carns
T. G. Eppink (w/o Attachments)
R. J. White
L. A. Reyes
NRC Resident Inspector
P. Ledbetter
K. M. Sutton
T. P. O'Kelley
RTS (LAR 02-3626)
File (813.20)
DMS (RC-03-0155)

**Subject: LICENSE AMENDMENT REQUEST - LAR 02-3626
REACTIVITY CONTROL SYSTEMS - MODERATOR TEMPERATURE COEFFICIENT**

## 1.0 DESCRIPTION

South Carolina Electric & Gas Company (SCE&G) requests an amendment to revise the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) Surveillance Requirements (SR). The proposed change will revise the near-end of life (EOL) Moderator Temperature Coefficient (MTC) Surveillance Requirement 4.1.1.3.b by placing a set of conditions on core operation, which if met, would allow exemption from the required MTC measurement. The conditional exemption will be determined on a cycle-specific basis by considering the margin predicted to the surveillance requirement MTC limit and the performance of core parameters, such as the beginning of life (BOL) MTC measurements and the critical boron concentration as a function of cycle length. The conditional exemption will improve plant availability and minimize disruptions to normal plant operations. Plant safety criteria will not be compromised by the conditional exemption of this one measurement. No changes to the TS Bases will be required as a result of the proposed amendment.

## 2.0 PROPOSED CHANGE

Specifically the proposed changes would revise the following:

2.1 TS 4.1.1.3.b

SR 4.1.1.3.b is revised to suspend the MTC measurement if the model benchmark criteria and Revised Prediction specified in the Core Operating Limits Report (COLR) are satisfied.

2.2 WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," is added to the list of references for the COLR in TS 6.9.1.11.

## 3.0 BACKGROUND

One of the controlling parameters for power and reactivity increases is the MTC. The requirements of TS 3.1.1.3 ensure that the MTC remains within the bounds used in the applicable Final Safety Analysis Report (FSAR) Chapter 15 accident analysis. This, in turn, ensures inherently stable power operations during normal operation and accident conditions.

The TS place both Limiting Condition for Operation (LCO) and SR constraints on the MTC, based on the accident analysis assumptions for the moderator density coefficient. A positive moderator density coefficient corresponds to a negative MTC. The most negative MTC LCO limit requires that the MTC be less negative that the specified limit for the all rods withdrawn, EOL, rated thermal power condition. To demonstrate compliance with the most negative MTC LCO, the surveillance requires verification of the MTC after 300 ppm equilibrium boron concentration is reached. Because the Hot Full Power (HFP) MTC value will gradually become more negative with further core burnup and boron concentration reduction, a 300 ppm MTC surveillance value should necessarily be less negative than the EOL LCO limit. To account for this effect, the 300 ppm MTC surveillance value is sufficiently less negative than the EOL LCO limit value, providing assurance that the LCO limit will be met as long as the 300 ppm MTC surveillance criterion is met.

Currently, the TS require measurements of MTC at BOL to verify the most positive MTC limit and near-EOL to verify the most negative MTC limit. At BOL, the measurement of the isothermal temperature coefficient is relatively simple to perform since it is done at hot zero power isothermal conditions and is not complicated by changes in the enthalpy rise or the presence of xenon. The measurement made near-EOL is performed at or near HFP conditions. MTC measurements at HFP are more difficult to perform due to:

- small variations in soluble boron concentration
- changes in xenon concentration and distribution
- changes in fuel temperature
- and changes in enthalpy rise

created by small changes in the core average power during the measurement. Changes in each of these parameters must be accurately accounted for when reducing the measurement data, or additional measurement uncertainties will be introduced. Even though these additional uncertainties may be small, the total reactivity change associated with the swing in moderator temperature is also relatively small. The resulting MTC measurement uncertainty created by even a small change in power level can then become significant and, if improperly accounted for, can yield misleading measurement results.

Each measurement of MTC requires several hours at less than full power operation (as a buffer to measurement-induced transients) and requires additional manpower. This presents a perturbation to normal operation and to the reactor itself. An alternate method is proposed for use at VCSNS to improve availability and minimize disruption to normal plant operations. The MTC measurement is replaced by a design calculation of the core MTC if predefined requirements are met.

The proposed change will allow modification of the EOL MTC surveillance requirement by placing a set of conditions on core operations. If these conditions are met, i.e., the specified revised prediction of the MTC and limits for several core parameters measured during the cycle are within specified bounds, the surveillance measurement would not be required.

## 4.0 TECHNICAL ANALYSIS

The conditional exemption from the HFP near-EOL 300 ppm MTC measurement does not impact the safe operation of VCSNS. The safety analysis assumption of a constant moderator density coefficient and the actual value assumed will not change. The TS Bases for and values of the most negative MTC limiting condition for operation and for the surveillance requirement are not altered. Instead, a revised prediction is compared to the surveillance MTC to determine if the limit is met. The method for calculating the revised prediction is consistent with the approved methodology of WCAP-13749-P-A (Reference 1 of this enclosure).

The methodology for the proposed change was submitted to the NRC as Westinghouse Topical Report WCAP-13749 in May 1993. In October 1996, the NRC determined the report to be acceptable for referencing in license applications to the extent specified and under the limitations stated in the Brookhaven technical evaluation report and the NRC staff's safety evaluation report. Reference 1 includes all of these documents.

The topical report was approved by the NRC with two requirements:

- only PHOENIX/ANC calculation methods are used for the individual plant analyses relevant to determinations for the EOL MTC plant methodology, and
- the predictive correction is reexamined if changes in core fuel designs or continued MTC calculation/measurement data show significant effect on the predictive correction.

VCSNS will meet both of these requirements. The PHOENIX/ANC calculation methods are used for VCSNS core designs. Prior to use of the conditional elimination technique, VCSNS will confirm that core design changes and MTC calculation and measurement data do not show a significant effect on the predictive correction. If a significant effect is found, the use of the predictive correction will be re-examined.

All of the core performance benchmark criteria, which are confirmed from startup physics test results, from routine HFP boron concentration measurements, and from flux map surveillances performed during the cycle, must be met before the Revised Predicted MTC can be calculated per the prescribed algorithm in Reference 1.

Enhancement

SCE&G is using NRC approved WCAP-13749-P-A as the basis for this license amendment request. SCE&G will meet all of the technical requirements in the approved WCAP, but proposes an enhancement to reduce regulatory burden for both the NRC and the licensee. SCE&G proposes not to submit a "Most Negative Moderator Temperature Coefficient Limit Report" (the Report) to the NRC. There are two reasons for this. First, there is an inconsistency in the WCAP regarding the time frame of data collection and the submittal of the Report to the NRC. More importantly, the Report serves no apparent technical or business need. Each of these reasons is explained below.

First, Section 3.3.3 of the WCAP states:

> "The Technical Specification Bases of the most negative MTC LCO and SR and the values of these limits are not altered. Instead, a revised prediction is compared to the SR MTC to determine if the SR limit is met. The revised prediction is simply the sum of the predicted HFP 300 ppm SR MTC plus and AFD correction factor plus a predictive correction term. This algorithm is summarized in Table 3-3."

Table D-2 of the WCAP states that the algorithm for determining the revised predicted near-EOL 300 ppm MTC is (emphasis added):

> "The Revised Predicted MTC = Predicted MTC + AFD Correction + Predicted Correction"
>
> where
>
> "Predicted MTC is calculated from Figure 1 (Predicted HFP ARO 300 ppm MTC Versus Cycle Burnup) *at the burnup corresponding to the measurement of 300 ppm* at RTP conditions...."

Table D-3 of the WCAP provides an example worksheet for calculating the revised predicted near-EOL 300 ppm MTC. Two of the required data inputs for the worksheet (B.1 and B.2) are used to calculate the AFD correction term in the algorithm (emphasis added):

B.1 Burnup of *most recent* HFP, equilibrium ____________
MWD/MTU
Conditions incore flux map

B.2 Measured HFP AFD at burnup (B.1) ____________
Reference incore flux map
I.D. ______________ Date: ___________

However, Appendix A to the WCAP requires a new TS 6.9.1.7 to be added (emphasis added):

> 6.9.1.7 The most negative MTC limits shall be provided to the NRC Regional Administrator with a copy to the Director of Nuclear Regulation, Attention: Chief, Core Performance Branch, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, *at least 60 days prior* to the date the limit would become effective unless otherwise approved by the Commission by letter. *This report will include the data required for the determination of the Revised Prediction* of the 300 ppm/ARO/RTP MTC per WCAP-13749, "Safety Evaluation Supporting the Conditional Elimination of the Most Negative EOL Moderator Temperature Coefficient Measurement", May, 1993 (Westinghouse Proprietary).

Because the Report would have to be submitted at least 60 days before reaching 300 ppm boron concentration, it cannot include the 300 ppm data required for determining the Revised Prediction. To meet the Report submittal requirement, the data to be used in calculating the revised predicted MTC may have to be taken 60 to 90 days prior to reaching 300 ppm boron. The WCAP does not provide any method for adjusting the revised predicted MTC to account for data collected 60 to 90 days prior to 300 ppm, nor does it provide justification for using such early data in the calculation. Therefore, the requirement to submit the Report and the requirements for the data that go into the report are inconsistent.

More importantly, the Report serves no apparent technical or business need. The applicability restrictions in the WCAP, the algorithm, and the acceptance criteria of the proposed Report would be included in the station procedure governing the EOL MTC surveillance. There is no compelling reason that this particular surveillance should require notifying the NRC prior to performing the surveillance procedure.

## 5.0 REGULATORY SAFETY ANALYSIS

### 5.1 No Significant Hazards Consideration

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?

Response: No.

The probability or consequences of accidents previously evaluated in the VCSNS FSAR are unaffected by this proposed change because there is no change to any equipment response or accident mitigation scenario. There are no additional challenges to fission product barrier integrity. Therefore, the proposed change does not involve a significant increase in the probability or 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?

Response: No.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed change does not challenge the performance or integrity of any safety-related system. Therefore, the proposed change does not create the possibility of a new or different knid of accident from any previously evaluated.

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

Response: No.

The margin of safety associated with the acceptance criteria of any accident is unchanged. The proposed change will have no affect on the availability, operability, or performance of the safety-related systems and components. A change to the surveillance requirement is proposed, but the limiting conditions for operation required by TS are not changed.

The TS Bases are founded in part on the ability of the regulatory criteria to be satisfied assuming the limiting conditions for operation are met for the various systems. Conformance to regulatory criteria for operation with the conditional exemption from the near-EOL MTC measurement is demonstrated and the regulatory limits are not exceeded. Therefore, the margin of safety as defined in the TS is not reduced and the proposed change does not involve a significant reduction in a margin of safety.

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

## 5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.36 (c) (3), "Surveillance Requirements" stipulates that surveillances be performed to assure the necessary quality of systems and components be maintained, the facility operation will be within safety limits, and that the limiting condition for operation will be met.

10 CFR 50 Appendix A, Criterion 1, "Quality Standards and Records," requires that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.

10 CFR 50 Appendix A, Criterion 10, "Reactor Design," requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

The proposed change does not violate any requirement or recommended method for assuring and maintaining the plant design and licensing basis.

5.2.1 Regulations

The regulatory basis for TS 4.1.1.3, "Moderator Temperature Coefficient," is to ensure that the value of the coefficient (MTC) remains within the limiting condition assumed in the VCSNS FSAR accident and transient analyses.

10 CFR Part 50, Appendix A (Reference 3), General Design Criterion (GDC) 11, "Reactor Inherent Protection," requires that the reactor core and associated coolant systems be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

GDC 12, "Suppression of Reactor Power Oscillations," requires a core design to assure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible.

5.2.2 Design Bases (FSAR)

FSAR Section 15.1.6 - REACTIVITY COEFFICIENTS

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients and their values are discussed in detail in Chapter 4.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Some analyses such as loss of reactor coolant from cracks or ruptures in the reactor coolant system do not depend on reactivity feedback effects. The values used are given in Table 15.1-4. Reference is made in that table to Figure 15.1-5 that shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. The justifications for use of conservatively large versus small reactivity coefficient values are treated on an event-by-event basis.

FSAR Section 15.2.7 - LOSS OF EXTERNAL ELECTRICAL LOAD AND/OR TURBINE TRIP

A major load loss on the plant can result from loss of external electrical load or from a turbine trip. For either case, offsite power is available for the continued operation of plant components such as the reactor coolant pumps. The case of loss of offsite power to the station auxiliaries is analyzed in Section 15.2.9.

The turbine trip is analyzed with both maximum and minimum reactivity feedback. The maximum feedback (EOL) cases assume a large negative moderator temperature coefficient and the most negative Doppler power coefficient. The minimum feedback (BOL) cases assume a minimum moderator temperature coefficient and the least negative Doppler coefficient.

FSAR Section 15.2.10 - EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTION

Excessive feedwater additions are a means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower and overtemperature protection (high neutron flux, overtemperature $\Delta T$, and overpower $\Delta T$ trips) prevent any power increase that could lead to a DNBR that is less than the DNBR limit.

FSAR Section 15.2.11 - EXCESSIVE LOAD INCREASE INCIDENT

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10% step load increase or a 5% per minute ramp load increase in the range of 15 to 100% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the Reactor Protection System. This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control.

For the BOL minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity and the least negative Doppler only power coefficient curve; therefore the least inherent transient response capability. For the EOL maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value and the most negative Doppler only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

FSAR Sub-Section 15.2.12.1 - Accidental Depressurization of the Reactor Coolant System

An accidental depressurization of the Reactor Coolant System could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flowrate of a relief valve, and will therefore allow a much more rapid depressurization upon opening, the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially, the event results in a rapidly decreasing RCS pressure, which could reach hot leg saturation conditions without Reactor Protection System intervention. If saturated conditions are reached, the rate of depressurization is slowed considerably. However, the pressure continues to decrease throughout the event. The effect of the pressure decrease is to increase power via moderator density feedback. However, if the plant is in the automatic mode, the rod control system functions to maintain the power essentially constant throughout the initial stages of the transient. The average coolant temperature remains approximately the same, but the pressurizer level increases until reactor trip because of the decreased reactor coolant density.

In calculating the DNBR the following conservative assumptions are made:

1. Plant characteristics and initial conditions are discussed in Section 15.1. Uncertainties and initial conditions are included in the limit DNBR.

2. A positive moderator temperature coefficient of reactivity for BOL operation in order to provide a conservatively high amount of positive reactivity feedback due to changes in moderator temperature. The spatial effect of voids due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape. These voids would tend to flatten the core power distribution.

3. A low (absolute value) *Doppler coefficient of reactivity such that the resultant* amount of negative feedback is conservatively low in order to maximize any power increase due to moderator reactivity feedback.

FSAR Section 15.2.13 - ACCIDENTAL DEPRESSURIZATION OF THE MAIN STEAM SYSTEM

FSAR Sub-Section 15.2.13.1 - Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the Main Steam System are associated with an inadvertent opening of a single steam dump, relief or safety valve. The analyses performed assuming a rupture of a main steam line are given in Section 15.4.

The steam release as a consequence of this accident results in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin.

The following conditions are assumed to exist at the time of a secondary steam system release.

1. End of life shutdown margin at no load, equilibrium xenon conditions, and with the most reactive rod cluster control assembly stuck in its fully withdrawn position. Operation of rod cluster control assembly banks during

core burnup is restricted in such a way that addition of positive reactivity in a secondary system steam release accident will not lead to a more adverse condition than the case analyzed.

2. A negative moderator coefficient corresponding to the end of life rodded core with the most reactive rod cluster control assembly in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included. The Keff versus temperature at 1150 psia corresponding to the negative moderator temperature coefficient plus the Doppler temperature effect used is shown in Figure 15.2-46.

FSAR Section 15.2.14 - INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM DURING POWER OPERATION

Spurious Emergency Core Cooling System (ECCS) operation at power could be caused by operator error or a false electrical actuating signal. A spurious signal in any of the following channels could cause this accident:

1. High containment pressure.
2. Low pressurizer pressure.
3. High steam line differential pressure.
4. Low steam line pressure.
5. Manual actuation.

FSAR Sub-Section 15.4.2.1 - Major Rupture of a Main Steam Line

The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the high power peaking factors which exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the safety injection system.

The limiting main steam line break was selected based upon the sensitivity studies performed in "Reactor Core Response to Excessive Secondary Steam Releases, "WCAP-9226, January, 1978 [42].

The analysis of a main steam line rupture is performed to demonstrate that, assuming a stuck RCCA (with and without offsite power), and a single failure in the engineered safety features (ESF) there is no consequential damage to the primary system and the core remains in place and intact.

FSAR Sub-Section 3.1.2.2 - Protection by Multiple Fission Product Barriers

Criterion 11 - Reactor Inherent Protection

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Discussion:

Prompt compensatory reactivity feedback effects are assured when the reactor is critical by the negative fuel temperature effect (Doppler effect) and by the nonpositive operational limit on the moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is assured by the inherent design using low enrichment fuel; the nonpositive moderator temperature coefficient of reactivity is assured by administratively controlling the dissolved absorber concentration or by burnable poison. These reactivity coefficients are discussed in Section 4.3.

Criterion 12 - Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible, or can be reliably and readily detected and suppressed.

Discussion:

Power oscillations of the fundamental mode are inherently eliminated by the negative Doppler and nonpositive moderator temperature coefficients of reactivity. Oscillations, due to xenon spatial effects, in the radial, diametral, and azimuthal overtone modes are heavily damped due to the inherent design and due to the negative Doppler and nonpositive moderator temperature coefficients of reactivity. Oscillations, due to xenon spatial effects, in the axial first overtone mode may occur. Assurance that fuel design limits are not exceeded by xenon axial oscillations is provided by reactor trip functions using the measured axial power imbalance as an input. Oscillations, due to xenon spatial effects, in axial modes higher than the first overtone, are heavily damped due to the inherent design and due to the negative Doppler coefficient of reactivity. Xenon stability control is discussed in Section 4.3.

5.2.3 Approved Methodologies

10 CFR 50 Appendix A, Criterion 10, "Reactor Design," requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

The regulatory basis for TS 4.1.1.3, "Moderator Temperature Coefficient," is to ensure that the value of the coefficient (MTC) remains within the limiting condition assumed in the VCSNS FSAR accident and transient analyses.

The proposed change will allow modification of the EOL MTC surveillance requirement by placing a set of conditions on core operations. If these conditions are met, i.e., the specified revised prediction of the MTC and limits for several core parameters measured during the cycle are within specified bounds, the surveillance measurement would not be required.

The methodology for the proposed change was submitted to the NRC as Westinghouse Topical Report WCAP-13749 in May 1993. In October 1996, the NRC determined the report to be acceptable for referencing in license applications to the extent specified and under the limitations stated in the Brookhaven technical evaluation report and the NRC staff's safety evaluation report.

5.2.4 Analysis

The conditional exemption from the HFP near-EOL 300 ppm MTC measurement does not impact the safe operation of VCSNS. The safety analysis assumption of a constant moderator density coefficient and the actual value assumed will not change. The TS Bases for and values of the most negative MTC limiting condition for operation and for the surveillance requirement are not altered. Instead, a revised prediction is compared to the surveillance MTC to determine if the limit is met. The method for calculating the revised prediction is consistent with the approved methodology of WCAP-13749-P-A (Reference 1 of this enclosure).

The analysis presented in FSAR Chapter 15 conforms to the acceptance criteria of 10 CFR 50, Appendix A.

5.2.5 Conclusion

The technical analysis performed by SCE&G demonstrates that the proposed amendment has no impact on core performance. Therefore, the proposed

License amendment is in compliance with GDC 10 and 11 as well as 10 CFR 50, Appendix A.

## 6.0 ENVIRONMENTAL CONSIDERATION

SCE&G has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20 (Reference 4), or would change an inspection or surveillance requirement. SCE&G has evaluated the proposed change and has determined that the change does not involve, (i) a significant hazards consideration, (ii) a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. As discussed above, the proposed changes do not involve a significant hazards consideration. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51 (Reference 5), specifically 10 CFR 51.22(c)(9). Therefore, pursuant 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

## 7.0 REFERENCES

1. Westinghouse Topical Report WCAP-13749-P, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL [end-of-life] Moderator Temperature Coefficient Measurement," March 1997

2. FSAR Section 15

3. 10 CFR 50, Appendix A, GDC

4. 10 CFR 20

5. 10 CFR 51

## ATTACHMENT I

## PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

**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 1-5 | 3/4 1-5 |
| 6-16a | 6-16a |

**SCE&G – EXPLANATION OF CHANGES**

| Page | Affected Section | Bar # | Description of Change | Reason for Change |
|---|---|---|---|---|
| 3/4 1-5 | 4.1.1.3.b | 1 | Insert * after first statement to denote footnote. | To identify footnote applicable to MTC surveillance. |
| | | 2 | Add footnote. | To provide qualifying statement for elimination of MTC measurement. |
| 6-16a | 6.9.1.11.a | 1 | Add "s" to "Limit" to title of 3.1.3.6. | To correct typo. |
| | | 2 | Insert "e". | To provide reference to WCAP-13749-P-A. |

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit specified in the COLR prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm.* In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

* Measurement of the MTC in accordance with SR 4.1.1.3.b may be suspended, provided that the benchmark criteria in WCAP-13749-P-A and the Revised Prediction specified in the COLR are satisfied.

CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limit, 3.1.3.6 - Control Rod Insertion Limit, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL $F_Q$ SURVEILLANCE TECHNICAL SPECIFICATION", February 1994 (W Proprietary).

(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor ($F_Q$ Methodology for W(z) surveillance requirements)).

c. WCAP-10266-P-A, Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987; including Addendum 2-A, "BASH METHODOLOGY IMPROVEMENTS AND RELIABILITY ENHANCEMENTS", May 1988, (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

d. WCAP-12472-P-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM", August 1994, (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.4 - Quadrant Power Tilt Ratio).

e. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement", March 1997, (Westinghouse Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements there to shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

**ATTACHMENT II**

**PROPOSED TECHNICAL SPECIFICATION CHANGES (RETYPED)**

## SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit specified in the COLR prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm*. In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

* Measurement of the MTC in accordance with SR 4.1.1.3.b may be suspended, provided that the benchmark criteria in WCAP-13749-P-A and the Revised Prediction specified in the COLR are satisfied.

CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Rod Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL $F_Q$ SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).

(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor ($F_Q$ Methodology for W(z) surveillance requirements).)

c. WCAP-10266-P-A, Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987; Including Addendum 2-A, "BASH METHODOLOGY IMPROVEMENTS AND RELIABILITY ENHANCEMENTS," May 1988, (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

d. WCAP-12472-P-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," August 1994, (W Proprietary).

(Methodology for Specifications 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.4 - Quadrant Power Tilt Ratio.)

e. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997, (Westinghouse Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements there to shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

## ATTACHMENT III

## LIST OF REGULATORY COMMITMENTS

1. The topical report was approved by the NRC with two requirements:

   - only PHOENIX/ANC calculation methods are used for the individual plant analyses relevant to determinations for the EOL MTC plant methodology, and
   - the predictive correction is reexamined if changes in core fuel designs or continued MTC calculation/measurement data show significant effect on the predictive correction.

   VCSNS will meet both of these requirements. The PHOENIX/ANC calculation methods are used for VCSNS core designs. Prior to use of the conditional elimination technique, VCSNS will confirm that core design changes and MTC calculation and measurement data do not show a significant effect on the predictive correction. If a significant effect is found, the use of the predictive correction will be re-examined.

2. All of the core performance benchmark criteria, which are confirmed from startup physics test results, from routine HFP boron concentration measurements, and from flux map surveillances performed during the cycle, must be met before the Revised Predicted MTC can be calculated per the prescribed algorithm in Reference 1.

## ATTACHMENT IV

## TYPICAL REVISED CORE OPERATING LIMITS REPORT PAGES

**(For Information Only)**

NUCLEAR OPERATIONS
COPY NO. 191

**SOUTH CAROLINA ELECTRIC & GAS COMPANY**

**VIRGIL C. SUMMER NUCLEAR STATION**

INFORMATION ONLY

# CORE OPERATING LIMITS REPORT

# FOR

# CYCLE 14

INFORMATION ONLY

**REVISION 0**

**APRIL 2002**

INFORMATION ONLY



## 2.0 Operating Limits

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the subsections which follow. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 6.9.1.11.

### 2.1 Moderator Temperature Coefficient (Specification 3.1.1.3):

2.1.1 The Moderator Temperature Coefficient (MTC) limits are:
The BOL/ARO-MTC shall be less positive than the limits shown in Figure 1.
The EOL/ARO/RTP-MTC shall be less negative than $-5x10^{-4}$ $\Delta k/k/°F$.

2.1.2 The MTC Surveillance limit is:
The 300 ppm/ARO/RTP-MTC should be less negative than or equal to $-4.1x10^{-4}$ $\Delta k/k/°F$
where: BOL stands for Beginning of Cycle Life
ARO stands for All Rods Out
RTP stands for RATED THERMAL POWER
EOL stands for End of Cycle Life

2.1.3 *INSERT 2.1.3

### 2.2 Shutdown Rod Insertion Limits (Specification 3.1.3.5):

The shutdown rods shall be withdrawn to at least 228 steps.

### 2.3 Control Rod Insertion Limits (Specification 3.1.3.6):

The Control Bank Insertion Limits are specified by Figure 2.

### 2.4 Axial Flux Difference (Specification 3.2.1):

2.4.1 The Axial Flux Difference (AFD) Limits for RAOC operation for Cycle 14 are shown in Figure 3.

2.4.2 The Axial Flux Difference (AFD) target band during base load operations for Cycle 14 is:
BOL - EOL (0 - 22,250 MWD/MTU): ±5% about a measured target value.

2.4.3 The minimum allowable power level for base load operation, $APL^{ND}$, is 85% of RATED THERMAL POWER.

**Insert 2.1.3**

2.1.3 The Revised Predicted near-EOL 300 ppm MTC shall be calculated using the following algorithm from Reference 3.2:

Revised Predicted MTC = Predicted MTC + AFD Correction + Predicted Correction*

* Predicted Correction is -3 pcm/°F.

If the Revised Predicted MTC is less negative than the SR 4.1.1.3.b limit of $-5.36 \times 10^{-4}$ Δk/k/°F, and all of the benchmark data contained in the surveillance procedure are met, then an MTC measurement in accordance with SR 4.1.1.3.b is not required.

If the Power Distribution Monitoring System is INOPERABLE, as defined in Technical Specification 3.3.3.11, the uncertainty, $U_{FQ}$, to be applied to the Heat Flux Hot Channel Factor $F_Q(z)$ shall be calculated by the following formula

$$U_{FQ} = U_{qu} \cdot U_e$$

where: $U_{qu}$ = Base $F_Q$ measurement uncertainty.
= 1.05
$U_e$ = Engineering uncertainty factor.
= 1.03

## 3.0 References


1) WCAP-12473-A (Non-Proprietary), "BEACON Core Monitoring and Operations Support System", August, 1994.

2) WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997, (W Proprietary)


INFORMATION ONLY