

March 6, 1990

Docket No. 50-395

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

Dear Mr. Bradham:

SUBJECT: ISSUANCE OF AMENDMENT NO. 88 TO FACILITY OPERATING LICENSE
NO. NPF-12 - VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1,
REGARDING DELETION OF CYCLE SPECIFIC PARAMETER FROM TECHNICAL
SPECIFICATION (TAC NO. 775049)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 88 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 (TS) in response to your application dated September 19, 1989, as supplemented October 19, 1989 and December 11, 1989.

Your September 19, 1989 submittal requested that the TS would be changed so that the values of cycle specific parameters would be deleted and reference made to a Core Operating Limits Report for the value of those parameters. The proposed changes are in accordance with recommendations of Generic Letter 80-16, "Removal of Cycle - Specific Parameter Limits from Technical Specifications." The amendment changes the TS and grants this request.

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

Sincerely,

Original Signed By:

9003230199 900306
FOR ADOCK 05000395
P FDC

John J. Hayes, Jr., Project Manager
Project Directorate II-1
Division of Reactor Projects I/II

Enclosures:

- 1. Amendment No. 88 to NPF-12
 - 2. Safety Evaluation
- cc w/enclosures:
See next page

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NAME	: PAnderson: : JHayes: sw : Flourigny	:	:	:
DATE	: 1/25/90 : 1/25/90 : 2/23/90	:	:	:

OFFICIAL RECORD COPY

Handwritten: CFI
DFU1

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South Carolina Electric & Gas Company

Virgil C. Summer Nuclear Station

cc:

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AMENDMENT NO. 88 TO FACILITY OPERATING LICENSE NO. NPF-12 - SUMMER, UNIT 1

Docket File

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cc: Licensee/Applicant Service List

DF01
11



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. 88
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 September 19, 1989, as supplemented October 19, 1989 and December 11, 1989, 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:

9003230201 900306
PDR ADDCK 05000395
P PIC

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 88, 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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 6, 1990

*SEE PREVIOUS CONCURRENCE

OFC	:LA:PD21:DRPR:PM:PD21:DRPR:*	OGC	:D:PD21:DRPR/	:	:
NAME	:PAnderson:	:JHes:sw	:MYoung	:EAdensam	Mts
DATE	:3/5/90	:3/5/90	:2/15/90	:3/5/90	:

ATTACHMENT TO LICENSE AMENDMENT NO. 88
TO FACILITY OPERATING LICENSE NO. NPF-12
DOCKET NO. 50-395

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

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DEFINITIONS

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

- a. All penetrations required to be closed during accident conditions are either:
 - 1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or
 - 2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.4.
- b. All equipment hatches are closed and sealed,
- c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,
- d. The containment leakage rates are within the limits of Specification 3.6.1.2, and
- e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9a The CORE OPERATING LIMITS REPORT (COLR) is the unit specific document that provides core operating limits for the current operating reload cycle. The cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be less than or equal to that shown in Figure 3.1-0:

APPLICABILITY: Beginning of Cycle Life (BOL) Limit - MODES 1 and 2* only#
End of Cycle Life (EOL) Limit - MODES 1, 2 and 3 only#

ACTION:

- a. With the MTC more positive than the BOL limit specified in the COLR above, operation in MODES 1 and 2 may proceed provided:
 1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.
 2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.
 3. In lieu of any other report required by Specification 6.9.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.
- b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0

#See Special Test Exception 3.10.3

REACTIVITY CONTROL SYSTEMS

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.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods which are inserted in the core shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

- a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.
- b. With more than one full length rod misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.
- c. With more than one full length rod inoperable due to a rod control urgent failure alarm or obvious electrical problem in the rod control system for greater than 72 hours, be in HOT STANDBY within the following 6 hours.
- d. With one full length rod inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:
 1. The rod is restored to OPERABLE status within the above alignment requirements, or
 2. The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT (COLR); the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or
 3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

*See Special Test Exceptions 3.10.2 and 3.10.3.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

- a. Restore the rod to within the limit specified in the COLR, or
- b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be within the insertion limit specified in the COLR.

- a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and
- b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.0

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR) figure entitled Rod Group Insertion Limits versus Thermal Power For Three Loop Operation.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

- a. Restore the control banks to within the limits within two hours, or
- b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or
- c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions 3.10.2 and 3.10.3

#With K_{eff} greater than or equal to 1.0.

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

- a. the allowed operational space as specified in the CORE OPERATING LIMITS REPORT (COLR) for Relaxed Axial Offset Control (RAOC) operation, or
- b. within the target band specified in the COLR about the target flux difference during base load operation.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

- a. For RAOC operation with the indicated AFD outside of the applicable limits specified in the COLR,
 1. Either restore the indicated AFD to within the COLR specified limits within 15 minutes, or
 2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal 55% of RATED THERMAL POWER within the next 4 hours.
- b. For Base Load operation above APL^{ND**} with the indicated AFD outside of the applicable target band about the target flux differences:
 1. Either restore the indicated AFD to within the COLR specified target band within 15 minutes, or
 2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue Base Load operation within 30 minutes.
- c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the applicable RAOC limits.

*See Special Test Exception 3.10.2

** APL^{ND} is the minimum allowable power level for base load operation and will be specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.11.

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(z)$ shall be limited by the following relationships:

$$F_Q(z) \leq \frac{[F_Q^{RTP}]}{P} [K(z)] \text{ for } P > 0.5$$

$$F_Q(z) \leq \left[\frac{F_Q^{RTP}}{0.5} \right] [K(z)] \text{ for } P \leq 0.5$$

where F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

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

$K(z)$ = the normalized $F_Q(z)$ for a given core height specified in the COLR.

APPLICABILITY: MODE 1.

ACTION.

With $F_Q(z)$ exceeding its limit:

- a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(z)$ exceeds the limit.
- b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(z)$ is demonstrated through incore mapping to be within its limit.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

- a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.
- b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.
- c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{P \times W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{W(z) \times 0.5} \text{ for } P > 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, $K(z)$ is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and $W(z)$ is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP} , $K(z)$ and $W(z)$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

- d. Measuring $F_Q^M(z)$ according to the following schedule:
 1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined, * or
 2. At least once per 31 Effective Full Power Days, whichever occurs first.

*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and power distribution map obtained.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) increasing since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

- (1) $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c. or
- (2) $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) is not increasing.

f. With the relationships specified in Specification 4.2.2.2c. above not being satisfied:

- (1) Calculate the maximum percent over the core height (z) that $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{ \left[\frac{\frac{F_Q^M(z) \times W(z)}{RTP}}{\frac{F_Q(z) \times K(z)}{P}} \right] - 1 \right\} \times 100 \text{ for } P \geq 0.5$$

$$\left\{ \left[\frac{\frac{F_Q^M(z) \times W(z)}{RTP}}{\frac{F_Q}{0.5} \times K(z)} \right] - 1 \right\} \times 100 \text{ for } P < 0.5$$

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

- (2) One of the following actions shall be taken:
- (a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the applicable AFD limits by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or
 - (b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or
 - (c) Verify that the requirements of Specification 4.2.2.3 for Base Load operation are satisfied and enter Base Load operation.
- g. The limits specified in Specifications 4.2.2.2c., 4.2.2.2e., and 4.2.2.2f. above are not applicable in the following core plane regions:
- 1. Lower core region from 0 to 15%, inclusive.
 - 2. Upper core region from 85 to 100%, inclusive.

4.2.2.3 Base Load operation is permitted at powers above APL^{ND} if the following conditions are satisfied:

- a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within applicable target band about the target flux difference) during this time period. Base Load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and F_Q surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BLQ} is defined as the minimum value of:

$$APL^{BL} = \frac{F_Q^{RTP} \times K(z)}{F_Q^M(z) \times W(z)_{BL}} \times 100\%$$

over the core height (z) where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is F_Q^{RTP} . $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transient encountered during base load operation. F_Q^{RTP} , $K(z)$, and $W(z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

- b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

4.2.2.4 During Base Load Operation $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

- a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND} .
- b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.
- c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z) \text{ for } P > APL^{ND}}{P \times W(z)_{BL}}$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$. The F_Q limit is F_Q^{RTP} .

P is the relative THERMAL POWER. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. F_Q^{RTP} , $K(z)$ and $W(z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

- d. Measuring $F_Q^M(z)$ in conjunction with target flux difference determination according to the following schedule:
 1. Prior to entering BASE LOAD operation after satisfying Section 4.2.2.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and
 2. At least once per 31 Effective Full Power Days.
- e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

over the core height (z) increasing since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1. $F_Q^M(z)$ shall be increased by 2 percent over that specified in 4.2.2.4.c, or
2. $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until 2 successive maps indicate that the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) is not increasing.

f. With the relationship specified in 4.2.2.4.c above not being satisfied, either of the following actions shall be taken:

1. Place core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied, and remeasure $F_Q^M(z)$, or
2. Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the maximum percent calculated over the core height (z) with the following expression:

$$\left\{ \left[\frac{F_Q^M(z) \times W(z)_{BL}}{Q} \right] - 1 \right\} \times 100 \text{ for } P \geq \text{APL}^{ND}$$
$$\left[\frac{\text{RTP}}{F_Q \times K(z)} \right]$$

g. The limits specified in 4.2.2.4.c, 4.2.2.4.e, and 4.2.2.4.f above are not applicable in the following core plane regions:

1. Lower core region 0 to 15 percent, inclusive.
2. Upper core region 85 to 100 percent, inclusive.

4.2.2.5 When $F_Q(z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

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POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation as specified in the CORE OPERATING LIMITS REPORT (COLR) figure entitled RCS Total Flow Rate Versus R For Three Loop Operation.

Where:

a.
$$R = \frac{F_{\Delta H}^N}{F_{\Delta H}^{RTP} [1.0 + PF_{\Delta H} (1.0 - P)]}$$
,

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

c. $F_{\Delta H}^N$ = Measured values of $F_{\Delta H}^N$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F_{\Delta H}^N$ shall be used to calculate R since the RCS Total Flow Rate Versus R figure in the COLR includes measurement uncertainties of 2.1% for flow and 4% for incore measurement of $F_{\Delta H}^N$, and

d. $F_{\Delta H}^{RTP}$ = The $F_{\Delta H}^N$ limit at RATED THERMAL POWER specified in the COLR.

e. $PF_{\Delta H}$ = The Power Factor Multiplier specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation specified in the COLR:

- a. Within 2 hours either:
1. Restore the combination of RCS total flow rate and R to within the above limits, or
 2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.
- b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION. (Continued)

- c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation specified in the COLR prior to exceeding the following THERMAL POWER levels:
 1. A nominal 50% of RATED THERMAL POWER,
 2. A nominal 75% of RATED THERMAL POWER, and
 3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of specified in the COLR.

- a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and
- b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation specified in the COLR at least once per 12 hours when the most recently obtained value of R obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by measurement at least once per 18 months.

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REACTIVITY CONTROL SYSTEMS

BASES

MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting End of Cycle Life (EOL) MTC value. The 300 ppm surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting EOL MTC value.

The surveillance requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, and 5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide the required SHUTDOWN

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the calculated DNBR in the core at or above the design limit during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

- $F_Q(z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;
- $F_{\Delta H}^N$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

The limits on AFD will be provided in the COLR per Technical Specification 6.9.1.11.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

POWER DISTRIBUTION LIMIT

BASES

AXIAL FLUX DIFFERENCE (Continued)

At power levels below APL^{ND} , the limits on AFD are defined in the COLR consistent with the Relaxed Axial Offset Control (RAOC) operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g., load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.

At power levels greater than APL^{ND} , two modes of operation are permissible; (1) RAOC, the AFD limit of which are defined in the COLR and (2) Base Load operation, which is defined as the maintenance of the AFD within PFLR specifications band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND} . However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(z)$ less than its limiting value. To allow operation at the maximum permissible power level the Base Load operating procedure restricts the indicated AFD to relatively small target band (as specified in the COLR) and power swings ($APL^{ND} \leq \text{power} \leq APL^{BL}$ or 100% Rated Thermal Power, whichever is lower). For Base Load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24-hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the Base Load procedure. After the waiting period extended Base Load operation is permissible.

The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are: (1) outside the allowed delta-I power operating space (for RAOC operation), or (2) outside the allowed delta-I target band (for Base Load operation). These alarms are active when power is greater than: (1) 50% of RATED THERMAL POWER (for RAOC operation), or (2) APL^{ND} (for Base Load operation). Penalty deviation minutes for Base Load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed.

POWER DISTRIBUTION LIMIT

BASES

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

- a. Control rods in a single group move together with no individual rod insertion differing by more than ± 13 steps, indicated, from the group demand position.
- b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.
- c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.
- d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F_{\Delta H}^N$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on the RCS Total Flow Rate Versus R figure in the CORE OPERATING LIMITS REPORT (COLR), RCS flow rate and power may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if core power is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F_{\Delta H}^N$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R, as calculated in 3.2.3 and used in the RCS Total Flow Rate Versus R figure in the COLR, accounts for $F_{\Delta H}^N$ less than or equal to the $F_{\Delta H}^{RTP}$ limit specified in the COLR. This value is used in the various accident analyses where $F_{\Delta H}^N$ influences parameters other than DNBR, e.g., peak clad temperature and thus is the maximum "as measured" value allowed.

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is more than sufficient to offset any rod bow penalty and transition core penalty. The remaining margin is available for plant design flexibility.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

POWER DISTRIBUTION LIMIT

BASES

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

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either RAOC or Base Load operation, $W(z)$ or $W(z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$ is met. $W(z)$ accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{BL}$ accounts for the more restrictive operating limits allowed by Base Load operation which result in less severe transient values. The $W(z)$ and $W(z)_{BL}$ functions described above for normal operation are specified in the CORE OPERATING LIMITS REPORT (COLR) per Specification 6.9.1.11.

When RCS flow rate and $F_{\Delta H}^N$ are measured, no additional allowances are necessary prior to comparison with the limits of the RCS Total Flow Rate Versus R figure in the COLR. Measurement errors of 2.1% for RCS total flow rate and 4% for $F_{\Delta H}^N$ have been allowed for in determining the limits of the RCS Total Flow Rate Versus R figure in the COLR.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation specified on the RCS Total Flow Rate Versus R figure in the COLR.

3/4.2.4 QUADRANT POWER TILT RATIO

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

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

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

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

POWER DISTRIBUTION LIMIT

BASES

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

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of in the core at or above the design limit throughout each analyzed transient.

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

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR in the core at or above the design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 300°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into, or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

ADMINISTRATIVE CONTROLS

- e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and
- f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the Process Control Program (PCP) made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORV's or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted as set forth in 6.5 above.

CORE OPERATING LIMITS REPORT

6.9.1.11 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

- a. Moderator Temperature Coefficient BOL and EOL Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
- b. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
- c. Control Bank Insertion Limits for Specification 3/4.1.3.6,
- d. Axial Flux Difference Limits, target band, and APL^{ND} for Specification 3/4.2.1,
- e. Heat Flux Hot Channel Factor, F_Q^{RTP} , $K(Z)$, $W(Z)$, APL^{ND} and $W(Z)_{BL}$ for Specification 3/4.2.2,
- f. Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, limits for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

- a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT

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

- b. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 (W Proprietary).

(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (FQ 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 (W Proprietary).

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

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.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 88 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 September 19, 1989 (Ref. 1), as amended by letter dated October 19, 1989 (Ref. 2) and superseded by letter dated December 11, 1989 (Ref. 3), South Carolina Electric & Gas Company (the licensee) proposed changes to the Technical Specifications (TS) for the Virgil C. Summer Nuclear Station, Unit No. 1. The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to a Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988 (Ref. 4).

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

- (1) The Definition section of the TS was modified to include a definition of the Core Operating Limits Report that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with an NRC approved methodology that maintains the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications. This change is acceptable.
- (2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

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- (a) Specification 3.1.1.3 and Surveillance Requirement 4.1.1.3
The moderator temperature coefficient (MTC) limits for this specification and for this surveillance requirement are specified in the COLR.
- (b) Specification 3.1.3.1
This Specification's action statement, 3.1.3.1.a.2, was revised to reference the control rod insertion limits specified in the COLR.
- (c) Specification 3.1.3.5 and Surveillance Requirement 4.1.3.5
The shutdown bank insertion limit for this specification and for this surveillance requirement is specified in the COLR.
- (d) Specification 3.1.3.6
The control bank insertion limits for this specification are specified in the COLR.
- (e) Specification 3.2.1
The axial flux difference limits, target bank, and the minimum allowable power level for base load operation (APL) for this specification are specified in the COLR.
- (f) Specification 3.2.2 and Surveillance Requirement 4.2.2
The total peaking factor (F_0) limit at rated thermal power, the normalized F_0 limit as a function of core height $K(z)$, the transient xenon effect on F_0 as a function of core height, and the minimum allowable power level for base load operation for this specification and for this surveillance requirement are specified in the COLR.
- (g) Specification 3.2.3 and Surveillance Requirement 4.2.3
The nuclear enthalpy rise hot channel factor (F-delta-H) limit at rated thermal power, and the power factor multiplier (PF-delta-H) for this specification and surveillance requirement are specified in the COLR.

In addition, the bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to the sections enumerated above and to their associated bases are consistent with the Generic Letter and thus, are acceptable.

- (3) Specification 6.9.1.11 is revised to delete a previous reporting requirement on Radial Peaking Factor Limit Report and to add the Core Operating Limits Report to the reporting requirements of the Administrative Controls section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, these specifications require that the values of these limits be established using NRC approved methodologies and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:
- (a) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (W Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limit, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)
 - (b) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification," June 1983 (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 (W Proprietary).

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

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in

the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

3.0 SUMMARY

The staff has reviewed the request by the South Carolina Electric & Gas Company to modify the Technical Specifications of the Virgil C. Summer Nuclear Station that would remove the specific values of some cycle-dependent parameters from the specifications and place the values in a Core Operating Limits Report that would be referenced by the Specification. Based on this review, the staff concludes that these TS modifications are acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, changes to the surveillance requirements, and changes to reporting 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) and (10). 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.

5.0 CONCLUSION

The Commission has issued a "Notice of Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing" which was published in the FEDERAL REGISTER on December 27, 1989 (54 FR 53211) 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Letter from O. S. Bradham (SCE&G) to NRC, dated September 19, 1989.
2. Letter from O. S. Bradham (SCE&G) to NRC, dated October 19, 1989.
3. Letter from O. S. Bradham (SCE&G) to NRC, dated December 11, 1989.
4. Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

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