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

March 26, 1991

Docket No. 50-400

Mr. Lynn W. Eury
Executive Vice President
Power Supply
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Eury:

SUBJECT: ISSUANCE OF AMENDMENT NO. 25 TO FACILITY OPERATING LICENSE NO. NPF-63 REGARDING PLACING CYCLE-SPECIFIC CORE OPERATING LIMITS AND RESTRICTIONS IN CORE OPERATING LIMITS REPORT FOR SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 (TAC NO. 77646)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 25 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1 (Harris). This amendment consists of changes to the Technical Specifications (TS) in response to your request dated September 19, 1990, as supplemented January 28, 1991, and February 18, 1991.

The amendment relocates certain numerical values for several cycle-specific core operating limits and restrictions from the Harris TS to the existing Harris Core Operating Limits Report. The cycle-specific parameters include: the moderator temperature coefficient, the heat flux hot channel factor, the normalized axial peaking factor, and the enthalpy rise hot channel factor. In addition, the amendment makes several administrative changes which clarify the definition and use of the term F sub-delta H, enthalpy rise hot channel factor, and provides descriptive phrases to enable plant personnel to better identify the location of information which has been removed from the TS and relocated to other plant documents.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's regular bi-weekly <u>Federal Register</u> notice.

Sincerely,

Ronnie Lo/for

Richard A. Becker, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 25 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PD21:DRPR	PM:PD21:DRPR	D:PD21:DRPR			
NAME	PAnderson	RBecker/dt	EAdensam			
DATE	2/20/90	2/22/90	2/12/90			

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AMENDMENT NO. 25 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

Docket File	
NRC PDR	
Local PDR	
PDII-1 Reading	
S. Varga	14 E4
G. Lainas	14 H3
E. Adensam	14 B20
P. Anderson	14 B20
R. Becker	14 B20
OGC	
D. Hagan	MNBB-3302
E. Jordan	MNBB-3302
G. Hill (4)	P1-137
Wanda Jones	P-130A
J. Calvo	11 D3
R. Jones	8 E23
T. Huang	8 E23
ACRS (10)	
GPA/PA	
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. NPF-63

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Carolina Power & Light Company (the licensee), dated September 19, 1990, as supplemented January 28, 1991, and February 18, 1991, 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-63 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 25, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Ronnie Lo for/

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 26, 1991

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	: OGC	:D:PD21:DRPR	:	:
NAME	:PAnderson	: RBecker	: E. Chan	:EAdensam	:	:
DATE	: 2/20/90	:2/21/90	: 3/14/90	:3/26/90	:	:

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ATTACHMENT TO LICENSE AMENDMENT NO. 25

FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
v	v
1-2	1-2
B 2-1a	B 2-1a
3/4 1-4	3/4 1-4
3/4 1-5	3/4 1-5
3/4 1-14	3/4 1-14
3/4 1-20	3/4 1-20
3/4 1-21	3/4 1-21
3/4 1-22	3/4 1-22
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B 3/4 1-2	B 3/4 1-2
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-2a	B 3/4 2-2a
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-	6-24b

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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 Specification 3.6.3.

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

1.9.a The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Plant operation within these core operating limits is addressed within the individual specifications.

DIGITAL CHANNEL OPERATIONAL TEST

1.10 A DIGITAL CHANNEL OPERATIONAL TEST shall consist of exercising the digital computer hardware using data base manipulation to verify OPERABILITY of alarm and/or trip functions.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE (Continued)

These curves are based on an enthalpy hot channel factor, $F_{\Delta H}$, specified in the CORE OPERATING LIMITS REPORT (COLR) and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in calculated $F_{\Delta H}$ at reduced power based on the expression:

$$F_{\Delta H} = F_{\Delta H}^{RTP} [1 + PF_{\Delta H} (1-P)]$$

Where P is the fraction of RATED THERMAL POWER,

$F_{\Delta H}^{RTP}$ = $F_{\Delta H}$ limit at RATED THERMAL POWER specified in the COLR, and

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

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106. The maximum positive limit shall be less than or equal to +5 pcm/°F for power levels up to 70% RATED THERMAL POWER and a linear ramp from that point to 0 pcm/°F at 100% RATED THERMAL POWER.

APPLICABILITY: Positive MTC Limit - MODES 1 and 2* only**.
Negative MTC Limit - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the Positive MTC Limit specified in the COLR, operation in MODES 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to within the Positive MTC 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; and

   3. 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 Negative MTC Limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*With k_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

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 Positive MTC Limit specified in the COLR, plant procedure PLP-106, prior to initial operation above 5% of RATED THERMAL POWER after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR 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 Negative MTC Limit specified in the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All shutdown and control rods 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 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 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 rod inoperable, due to a rod control urgent failure alarm or obvious electrical problem in the rod control system existing for greater than 36 hours, be in HOT STANDBY within the following 6 hours.

d. With one rod trippable but 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 1 hour:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. 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:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents

*See Special Test Exceptions Specifications 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 fully withdrawn as specified in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With a maximum of one shutdown rod not fully withdrawn as specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, 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 fully withdrawn as specified in the COLR:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.
**With K_{eff} greater than or equal to 1.

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), plant procedure PLP-106.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With the control banks inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the insertion limit specified in the COLR within 2 hours, or

b. Reduce THERMAL POWER within 2 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 limit specified in the COLR 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 Specifications 3.10.2 and 3.10.3.

**With K_{eff} greater than or equal to 1.

FIGURE 3.1-2

ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER, THREE LOOP OPERATION

This figure is deleted from Technical Specifications, and is controlled by the CORE OPERATING LIMITS REPORT, plant procedure PLP-106.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

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

a. the acceptable operational space as specified in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106, for Relaxed Axial Offset Control (RAOC) operation, or

b. within a band about the target AFD during Base Load operation as specified in the COLR.

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

ACTION:

a. For RAOC operation with the indicated AFD outside of the limits specified in the COLR, either:

   1. Restore the indicated AFD to within the limits specified in the COLR 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 to 55% of RATED THERMAL POWER within the next 4 hours.

b. For Base Load operation above APL^{ND}** with the indicated AXIAL FLUX DIFFERENCE outside of the band about the target AFD, either:

   1. Restore the indicated AFD to within the target band limits 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 limits specified in the COLR for RAOC operation.

*See Special Test Exception 3.10.2

**APL^{ND} is the minimum allowable power level for Base Load operation and is specified in the COLR.

FIGURE 3.2-1

<u>AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER FOR RAOC</u>

This figure is deleted from Technical Specifications and is controlled by the CORE OPERATING LIMITS REPORT, plant procedure PLP-106.

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:

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} \times K(Z) \text{ FOR } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} \times K(Z) \text{ FOR } P \leq 0.5$$

Where:

F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106,

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

K(Z) = the normalized $F_Q(Z)$ as a function of 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 ΔT 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 ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

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 it 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:

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

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

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 fraction of RATED 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 COLR.

d. Measuring $F_Q^M(Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding oy 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 a power distribution map obtained.

SURVEILLANCE REQUIREMENTS (Continued)

e. With measurements indicating

$$\text{maximum} \left(\frac{F_Q^M(Z)}{K(Z)} \right)$$

has increased 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

$$\text{maximum} \left(\frac{F_Q^M(Z)}{K(Z)} \right) \text{ is not increasing.}$$

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

1) Calculate the percent $F_Q(Z)$ exceeds its limit by the following expression:

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

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

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 AFD limits specified in the COLR 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 the limits specified in the Core Operating Limits Report) 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 FQ surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

$$APL^{BL} = \text{minimum} \left[\frac{F_Q^{RTP} \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}} \right] \times 100\%$$

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, and $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during Base Load operation. F_Q^{RTP}, K(Z), and $W(Z)_{BL}$ are specified in the COLR.

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 it 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:

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

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$, 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 fraction of RATED THERMAL POWER and $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during Base Load operation. F_Q^{RTP}, $K(Z)$, and $W(Z)_{BL}$ are specified in the COLR.

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 measurements indicating

$$\text{maximum} \quad \left[\frac{F_Q^M(Z)}{K(Z)}\right]$$

has increased since the previous determination $F_Q^M(Z)$ 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 EFPD until 2 successive maps indicate that

$$\text{maximum} \quad \left[\frac{F_Q^M(Z)}{K(Z)}\right] \quad \text{is not increasing}$$

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 the core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied, and remeasure $F_Q^M(Z)$, or

FIGURE 3.2-2

K(Z) - THE NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

This figure is deleted from Technical Specifications and is controlled by the CORE OPERATING LIMITS REPORT, plant procedure PLP-106.

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

LIMITING CONDITION FOR OPERATION

3.2.3 The Reactor Coolant System (RCS) total flow rate and $F_{\Delta H}$ shall be maintained as follows:

a. RCS total flow rate $\geq$ 293,540 gpm x $(1.0 + C_1)$, and

b. $F_{\Delta H} \leq F_{\Delta H}^{RTP}\ [1.0 + PF_{\Delta H}\ (1.0-P)]$

Where:

$F_{\Delta H}^{RTP}$ = $F_{\Delta H}$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106,

$PF_{\Delta H}$ = Power Factor Multipler for $F_{\Delta H}$ specified in the COLR,

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

$F_{\Delta H}$ = Enthalpy rise hot channel factor obtained by using the movable incore detectors to obtain a power distribution map, with the measured value of the nuclear enthalpy rise hot channel factor $(F_{\Delta H}^{N})$ increased by an allowance of 4% to account for measurement uncertainty, and

C_1 = Measurement uncertainty for core flow as described in the Bases.

APPLICABILITY: MODE 1.

ACTION:

With RCS total flow rate or $F_{\Delta H}$ outside the above limits:

a. Within 2 hours either:

1. Restore RCS total flow rate and $F_{\Delta H}$ 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.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate determination that $F_{\Delta H}$ and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate determination, to be within acceptable limits 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 $F_{\Delta H}$ shall be determined to be within acceptable limits:

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 RCS total flow rate shall be verified to be within the acceptable limit:

a. At least once per 12 hours by the use of main control board instrumentation or equivalent, and

b. At least once per 31 days by the use of process computer readings or digital voltmeter measurement.

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 precision heat balance measurement at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION: As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channels and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit, specified in the Technical Specification Equipment List Program, plant procedure PLP-106, at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-2 REACTOR TRIP SYSTEM INSTRUMENTATION
RESPONSE TIMES

This table is deleted from Technical Specifications.

The information in this table is controlled by the Technical Specification Equipment List Program, plant procedure PLP-106.

PAGE 3/4 3-10 HAS BEEN DELETED.

INSTRUCTIONS

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within its limit specified in the Technical Specification Equipment List Program, plant procedure PLP-106, at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

TABLE 3.3-5 ENGINEERED SAFETY FEATURES RESPONSE TIMES

This table is deleted from Technical Specifications.

The information in this table is controlled by the Technical Specification Equipment List Program, plant procedure PLP-106.

PAGES 3/4 3-38 THROUGH 3/4 3-40 HAVE BEEN DELETED.

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.2 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate as shown on Table 4.4-6.

b. A maximum cooldown rate as shown on Table 4.4-6.

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 4, 5, and 6 with reactor vessel head on.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; if the pressure and temperature limit lines shown on Figures 3.4-2 and 3.4-3 were exceeded, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or maintain the RCS T_{avg} and pressure at less than 200°F and 500 psig, respectively.

SURVEILLANCE REQUIREMENTS

4.4.9.2.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.2.2 Deleted from Technical Specifications. Refer to the Technical Specification Equipment List Program, plant procedure PLP-106.

TABLE 4.4-5 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

This table is deleted from Technical Specifications.

The information in this table is controlled by the Technical Specification Equipment List Program, plant procedure PLP-106.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve specified in the Technical Specification Equipment List Program, plant procedure PLP-106, shall be OPERABLE with isolation times less than or equal to required isolation times.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the containment isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Ventilation Isolation test signal, each normal, preentry purge makeup and exhaust, and containment vacuum relief valve actuates to its isolation position, and

d. Verifying that, on a Safety Injection "S" test signal, each containment isolation valve receiving an "S" signal actuates to its isolation position, and

e. Verifying that, on a Main Steam Isolation test signal, each main steam isolation valve actuates to its isolation position, and

f. Verifying that, on a Main Feedwater Isolation test signal, each feedwater isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit specified in the Technical Specification Equipment List Program, plant procedure PLP-106, when tested pursuant to Specification 4.0.5.

TABLE 3.6-1 CONTAINMENT ISOLATION VALVES

This table is deleted from Technical Specifications.

The information in this table is controlled by the Technical Specification Equipment List Program, plant procedure PLP-106.

PAGES 3/4 6-17 THROUGH 3/4 6-29 HAVE BEEN DELETED.

SURVEILLANCE REQUIREMENTS (Continued)

2. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the percent calculated with the following expression:

$$[(\text{max. of } [\frac{F_Q^M(Z) \times W(Z)_{BL}}{\frac{F_Q^{RTP}}{P} \times K(Z)}]) - 1] \times 100 \text{ for } P \geq APL^{ND}$$

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.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per the augmented inservice inspection program on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the augmented inservice inspection program specified in the Technical Specification Equipment List Program, plant procedure PLP-106.

PAGES 3/4 7-20 THROUGH 3/4 7-23 HAVE BEEN DELETED.

<u>FIGURE 4.7-1 SAMPLE PLAN (2) FOR SNUBBER FUNCTIONAL TEST</u>

This figure is deleted from Technical Specifications and is controlled by the Technical Specification Equipment List Program, plant procedure PLP-106.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 Each containment penetration conductor overcurrent protective device specified in the Technical Specification Equipment List Program, plant procedure PLP-106, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out or removed at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out or removed, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the 6900-volt circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers, and performing the following:

   a) A CHANNEL CALIBRATION of the associated protective relays,

   b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed, and

TABLE 3.8-1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

This table is deleted from Technical Specifications.

The information in this table is controlled by the Technical Specification Equipment List Program, plant procedure PLP-106.

PAGES 3/4 8-22 THROUGH 3/4 8-38B HAVE BEEN DELETED.

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve, specified in the Technical Specification Equipment List Program, plant procedure PLP-106, requiring bypass protection, shall be bypassed only under accident conditions by an OPERABLE bypass device integral with the motor starter.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not capable of being bypassed under conditions for which it is designed to be bypassed, restore the inoperable device or provide a means to bypass the thermal overload within 8 hours, or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) of the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2 The thermal overload protection for the above required valves shall be verified to be bypassed only under accident conditions by an OPERABLE integral bypass device by the performance of a TRIP ACTUATION DEVICE OPERATIONAL TEST of the bypass circuitry:

a. At least once per 18 months for those thermal overloads which are normally in force during plant operation and are bypassed only under accident conditions; and

b. Following maintenance on the motor starter.

TABLE 3.8-2 MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

This table is deleted from Technical Specifications.

The information in this table is controlled by the Technical Specification Equipment List Program, plant procedure PLP-106.

PAGES 3/4 8-41 THROUGH 3/4 8-43 HAVE BEEN DELETED.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR safety analyses to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) subtracting from this value the largest differences in MTC observed between EOL, all rods withdrawn, RATED THERMAL POWER conditions, and those most adverse conditions of moderator temperature and pressure, rod insertion, axial power skewing, and xenon concentration that can occur in normal operation and lead to a significantly more negative EOL MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR safety analyses into the Negative MTC Limit. The 300 ppm surveillance limit MTC value represents a conservative MTC value at a core condition of 300 ppm equilibrium boron concentration, and is obtained by making corrections for burnup and soluble boron to the Negative MTC Limit.

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 trip 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 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/safety injection 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 350°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 MARGIN as defined by Specification 3/4.1.1.2 after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at BOL

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 minimum DNBR in the core greater than or equal to the design DNBR value 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;

$F^N_{\Delta H}$ 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;

$F_{\Delta H}$ 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, with an allowance to account for measurement uncertainty.

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 normalized axial peaking factor is specified in the COLR.

Target flux difference (TARGET AFD) is determined at equilibrium xenon conditions. The 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 AFD at RATED THERMAL POWER for the associated core burnup conditions. TARGET AFD 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.

AXIAL FLUX DIFFERENCE (Continued)

At power levels below APL^{ND}, the limits on AFD are specified in the COLR for RAOC operation. 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 with fixed AFD limits as a function of reactor power level and 2) Base Load operation which is defined as the maintenance of the AFD within a band about a target value. Both the fixed AFD limits for RAOC operation and the band for Base Load operation are specified in the COLR. RAOC operations above APL^{ND} are the same as 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 value, the Base Load operating procedure restricts the indicated AFD to a relatively small target band and power swings. 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 two or more OPERABLE excore channels are: 1) outside the allowed ΔI power operating space (for RAOC operation), or 2) outside the acceptable AFD 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.

3/4.2.2 AND 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, RCS FLOW RATE, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flow rate, and 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 ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

POWER DISTRIBUTION LIMITS

BASES

HEAT FLUX HOT CHANNEL FACTOR, AND RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The combinations of the RCS flow requirement and the measurement of $F_{\Delta H}$ ensures that the calculated DNBR will not be below the design DNBR value. The relaxation of $F_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an $F_{\Delta H}$ measurement is taken, an allowance for measurement error must be applied prior to comparing to the $F_{\Delta H}^{RTP}$ limit(s) specified in the CORE OPERATING LIMITS REPORT (COLR). An allowance of 4% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System.

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.

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.

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 are specified in the COLR.

HEAT FLUX HOT CHANNEL FACTOR, AND RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

When RCS flow rate is measured, no additional allowance is necessary prior to comparison with the limit of Specification 3.2.3. A normal RCS flowrate error of 2.1% will be included in C_1, which will be modified as discussed below.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi, raises the nominal flow measurement allowance, C_1, to 2.2% for no venturi fouling. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation that could lead to operation outside the acceptable region of operation.

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. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-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% for each percent of tilt in excess of 1.

6.9.1.6 CORE OPERATING LIMITS REPORT

6.9.1.6.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106, prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. Moderator Temperature Coefficient Positive and Negative Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

b. Shutdown Bank Insertion Limits 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. Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

6.9.1.6.2 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).

(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-11914, "SAFETY EVALUATION SUPPORTING A MORE NEGATIVE EOL MODERATOR TEMPERATURE COEFFICIENT TECHNICAL SPECIFICATION FOR THE SHEARON HARRIS NUCLEAR POWER PLANT", August 1988 (W Proprietary). Approved by NRC Safety Evaluation dated May 22, 1989.

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient).

c. 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)).

d. WCAP-10266-P-A, Rev. 2, "The 1981 Version of the WESTINGHOUSE ECCS EVALUATION MODEL USING THE BASH CODE", March 1987 (W Proprietary).

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

e. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary).

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control)).

f. WCAP-11837-P-A, "EXTENSION OF METHODOLOGY FOR CALCULATING TRANSITION CORE DNBR PENALTIES", January 1990 (W Proprietary).

6.9.1.6.3 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.

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10CFR50.4 within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

RECORD RETENTION (Continued)

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 25 TO FACILITY OPERATING LICENSE NO. NPF-63

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By letter dated September 19, 1990 (Reference 1), as supplemented by letters dated January 28, 1991 (Reference 2), and February 18, 1991 (Reference 4), Carolina Power & Light Company (CP&L or the licensee) proposed changes to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (Harris). The proposed changes would modify additional specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to the Core Operating Limits Report (COLR) for the values of those limits. The use of the COLR for Harris was previously approved by the Nuclear Regulatory Commission (NRC) in Amendment No. 15, dated October 18, 1989. Guidance on the proposed changes was developed by the 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 (Reference 3). The January 28, 1991, and February 18, 1991, letters provided clarifying information that did not change the initial determination of no significant hazards consideration as published in the Federal Register.

2.0 EVALUATION

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

A. In addition to the approved cycle-specific core operating limits, dated October 18, 1989, the following specifications will be revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(1) 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 will be specified in the COLR.

(2) Specification 3.2.2 and Surveillance Requirement 4.2.2

The heat flux hot channel factor (F_Q) limit at rated thermal power, the normalized F_Q limit as a function of core height K(Z) and the transient effect on F_Q as a function of core height W(Z) for this specification and for this surveillance requirement are specified in the COLR.

(3) Specification 3.2.3 and Surveillance Requirement 4.2.3

The 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.

B. Specification 6.9.1.6. Core Operating Limits Report of the Administrative Controls section of the TS is revised to include currently proposed TS changes in Specification 6.9.1.6.1 and to add additional NRC-approved methodologies in Specification 6.9.1.6.2 to support the values of cycle-specific parameter limits that are applicable for the current fuel cycle. The approved methodologies are the following:

(1) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (Proprietary). This was in the approved COLR dated October 18, 1989.

(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.)

(2) WCAP-11914, "Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the Shearon Harris Nuclear Power Plant," August 1988 (Proprietary), approved by NRC safety evaluation dated May 23, 1989.

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

(3) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification," June 1983 (Proprietary). This was in the approved COLR dated October 18, 1989.

(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.)

(4) WCAP-10266-P-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987 (Proprietary).

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

(5) WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," September 1974 (Proprietary).

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

(6) WCAP-11837-P-A, "Extension of Methodology for Calculating Transition Core DNBR Penalties," January 1990 (Proprietary).

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.

On the basis of the review of the above items, the staff concludes that the licensee provided an acceptable response to those items addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in the TS. In addition, the licensee made a number of editorial clarifications of items that had been deleted or removed to other plant documents. 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 these changes are administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

3.0 SUMMARY

The staff has reviewed the request by the Carolina Power & Light Company to modify the Harris TS 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 specifications. Based on this review, the staff concludes that these TS modifications are acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

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

5.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the Federal Register (55 FR 42093) on October 17, 1990, and consulted with the State of North Carolina. No public comments or requests for hearing were received, and the State of North Carolina did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Letter (NLS-90-187) from A. B. Cutter (CP&L) to NRC, dated September 19, 1990.

2. Letter (NLS-91-018) from G. E. Vaughn (CP&L) to NRC, dated January 28, 1991.

3. Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

4. Letter (NLS-91-049) from S. D. Floyd (CP&L) to NRC, dated February 18, 1991.

Dated: March 26, 1991

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