August 16, 1988

Docket No. 50-400

DISTRIBUTION
See attached list

Mr. E. E. Utley
Senior Executive Vice President
Power Supply and Engineering & Construction
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

SUBJECT: ISSUANCE OF AMENDMENT NO. 7 TO FACILITY OPERATING LICENSE NO. NPF-63 - SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1, REGARDING TECHNICAL SPECIFICATIONS RELATING TO CYCLE 2 OPERATION (TAC NO. 67089)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 7 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1. This amendment consists of changes to the Technical Specifications in response to your request dated February 1 and amended February 8, 1988.

The amendment makes changes to control rod Bank-D configuration, increases the radial and total peaking factors $F_{\Delta H}$ and F_Q, introduces boron dilution/sliding shutdown margin and changes some miscellaneous technical specifications detailed in the Safety Evaluation. The changes are needed so that you can use higher enrichment fuel (up to 4.2 weight percent uranium 235), extend the fuel irradiation limits, and permit operation of longer fuel cycles. The environmental consequences of the use of higher enrichment fuel and extended irradiation were addressed in licensee submittals associated with Amendment 5, dated May 26, and November 2, 1987.

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

Sincerely,

/S/

Bart C. Buckley, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 7 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:D:PD21:DRPR	:	:	:
NAME	:PAnderson:	: BBuckley:ch:	EAdensam	:	:	:
DATE	: 8/15/88	: 8/15/8	: 8/16/88	:	:	:

OFFICIAL RECORD COPY

DFOL

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Carolina Power & Light Company

Shearon Harris

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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. 7
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 February 1 and amended February 8, 1988, 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. 7, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 16, 1988

OFC	LA:PD21:DRPR	PM:PD21:DRPR	OGC	D:PD21:DRPR			
NAME	PAnderson	BBuckley:ch		EAdensam			
DATE	8/15/88	8/12/88	8/ /88	8/16/88			

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ATTACHMENT TO LICENSE AMENDMENT NO.

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
iv	iv
v	v
xiii	xiii
2-2	2-2
2-8	2-8
B 2-1	B 2-1
3/4 1-1	3/4 1-1
3/4 1-2	3/4 1-2
3/4 1-3	3/4 1-3
-	3/4 1-3a
3/4 1-8	3/4 1-8
3/4 1-10	3/4 1-10
3/4 1-11	3/4 1-11
3/4 1-12	3/4 1-12
3/4 1-14	3/4 1-14
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3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
-	3/4 2-7a
-	3/4 2-7b
-	3/4 2-7c
-	3/4 2-7d
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 3-11	3/4 3-11
3/4 3-14	3/4 3-14
-	3/4 3-14a
B 3/4 1-1	B 3/4 1-1
-	B 3/4 1-1a
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
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 2-3	B 3/4 2-3
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-	5-6a
6-24	6-24
-	6-24a

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3.0/4.0 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

3.0/4.0 BASES

SECTION | PAGE

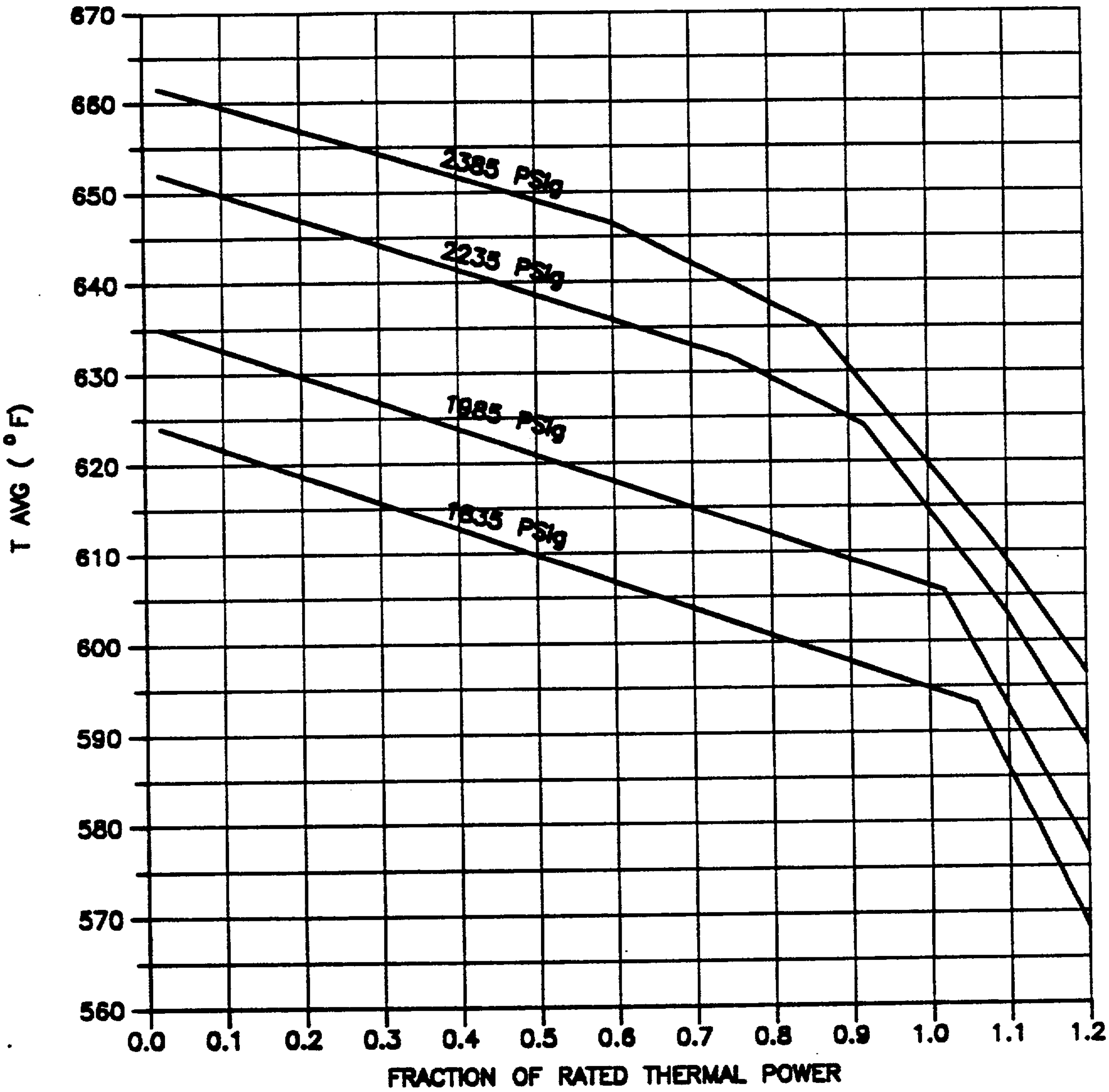


FIGURE 2.1-1
REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: (Continued)

T'	$\leq$ 588.8°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	= 0.000828/psig;
P	= Pressurizer pressure, psig;
P'	= 2235 psig (Nominal RCS operating pressure);
S	= Laplace transform operator, s^{-1};

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between -25% and + 7.0%, f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -25%, the ΔT Trip Setpoint shall be automatically reduced by 2.36% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds +7.0%, the ΔT Trip Setpoint shall be automatically reduced by 1.70% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.9% ΔT span.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the calculated heat flux that would cause DNB at a particular core location to the actual local heat flux and is indicative of the margin to DNB.

The minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in calculated $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1 + 0.3\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

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.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1770 pcm for 3-loop operation.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the SHUTDOWN MARGIN less than 1770 pcm, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1770 pcm:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. Within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

1) Reactor Coolant System boron concentration,
2) Control rod position,
3) Reactor Coolant System average temperature,
4) Fuel burnup based on gross thermal energy generation,
5) Xenon concentration, and
6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ±1000 pcm at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. If later experience shows adjustment is desirable at approximately 60 EFPD, the adjustment is permissible.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - MODES 3, 4, AND 5

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limit shown in Figure 3.1-1.

APPLICABILITY: MODES 3, 4, AND 5.

ACTION:

With the SHUTDOWN MARGIN less than the required value immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

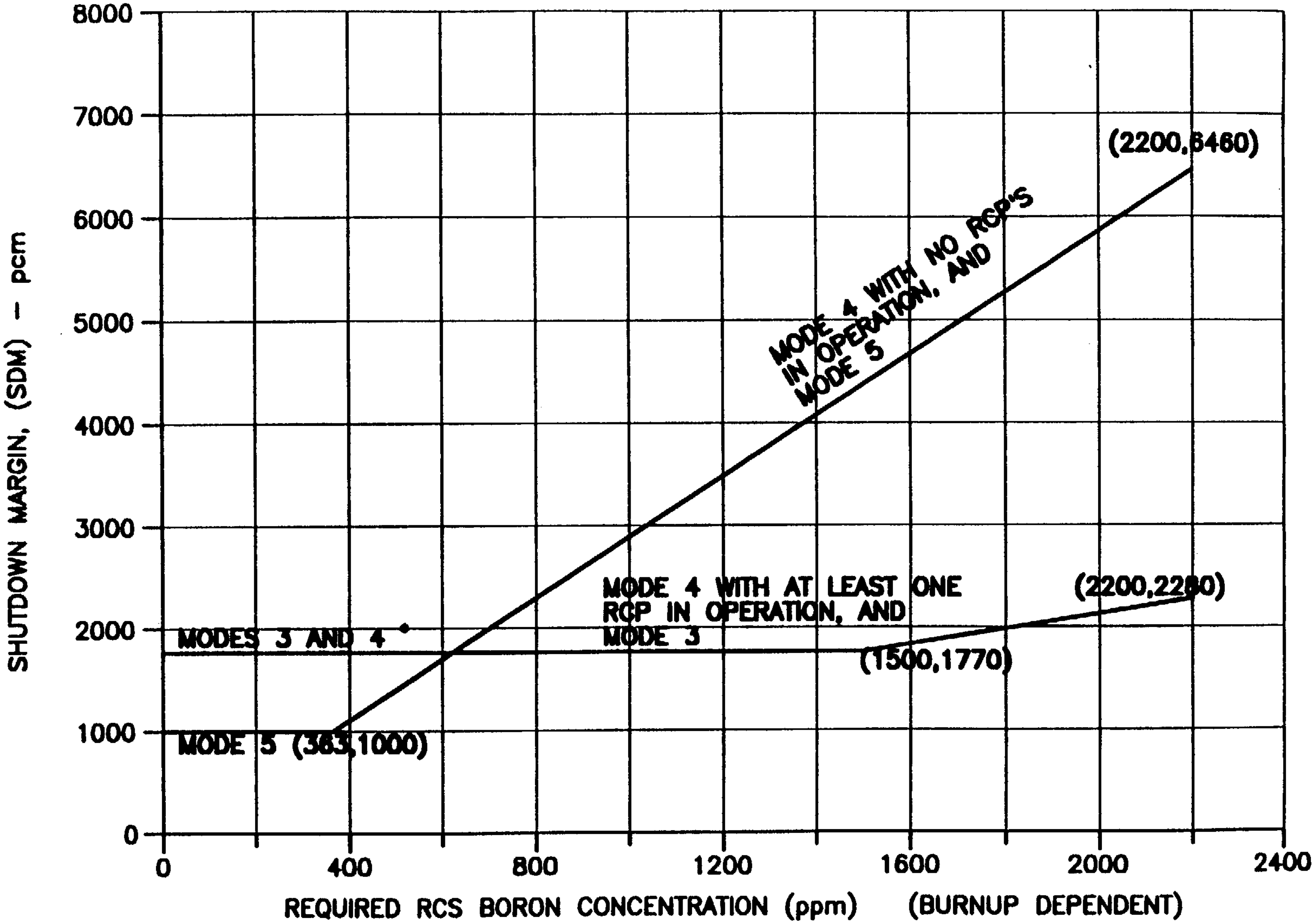
SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the required value:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

   1) Reactor Coolant System boron concentration,
   2) Control rod position,
   3) Reactor Coolant System average temperature,
   4) Fuel burnup based on gross thermal energy generation,
   5) Xenon concentration, and
   6) Samarium concentration.



* Applicable to Mode 4, with or without RCP's in operation

FIGURE 3.1-1
SHUTDOWN MARGIN VERSUS RCS BORON CONCENTRATION
MODES 3, 4, AND 5

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tank via a boric acid transfer pump and a charging/safety injection pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via charging/ safety injection pumps to the RCS.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as required by Figure 3.1-1 at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the flow path between the boric acid tank and the charging/safety injection pump suction header tank is greater than or equal to 65°F when a flow path from the boric acid tank is used;

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal; and

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the RCS.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging/safety injection pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one charging/safety injection pump OPERABLE, restore at least two charging/safety injection pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as required by Figure 3.1-1 at 200°F within the next 6 hours; restore at least two charging/safety injection pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging/safety injection pumps shall be demonstrated OPERABLE by verifying, on recirculation flow or in service supplying flow to the Reactor Coolant System and reactor coolant pump seals, that a differential pressure across each pump of greater than or equal to 2446 psid is developed when tested pursuant to Specification 4.0.5.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid tank with:

1. A minimum contained borated water volume of 7100 gallons, which is equivalent to 17% indicated level,
2. A boron concentration of between 7000 and 7750 ppm, and
3. A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1. A minimum contained borated water volume of 106,000 gallons, which is equivalent to 12% indicated level,
2. A boron concentration of between 2000 and 2200 ppm, and
3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the contained borated water volume, and
3. Verifying the boric acid tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 40°F.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A boric acid tank with:

1. A minimum contained borated water volume of 21,400 gallons, which is equivalent to 60% indicated level.

2. A boron concentration of between 7000 and 7750 ppm, and

3. A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1. A minimum contained borated water volume of 436,000 gallons, which is equivalent to 92% indicated level.

2. A boron concentration of between 2000 and 2200 ppm,

3. A minimum solution temperature of 40°F, and

4. A maximum solution temperature of 125°F.

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

ACTION:

a. With the boric acid tank inoperable and being used as one of the above required borated water sources, restore the boric acid tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN as required by Figure 3.1-1 at 200°F; restore the boric acid tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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 Figure 3.1-2. 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

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-2.

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:

a. Restore the control banks to within the limits 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 Figure 3.1-2, 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 Specifications 3.10.2 and 3.10.3.

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

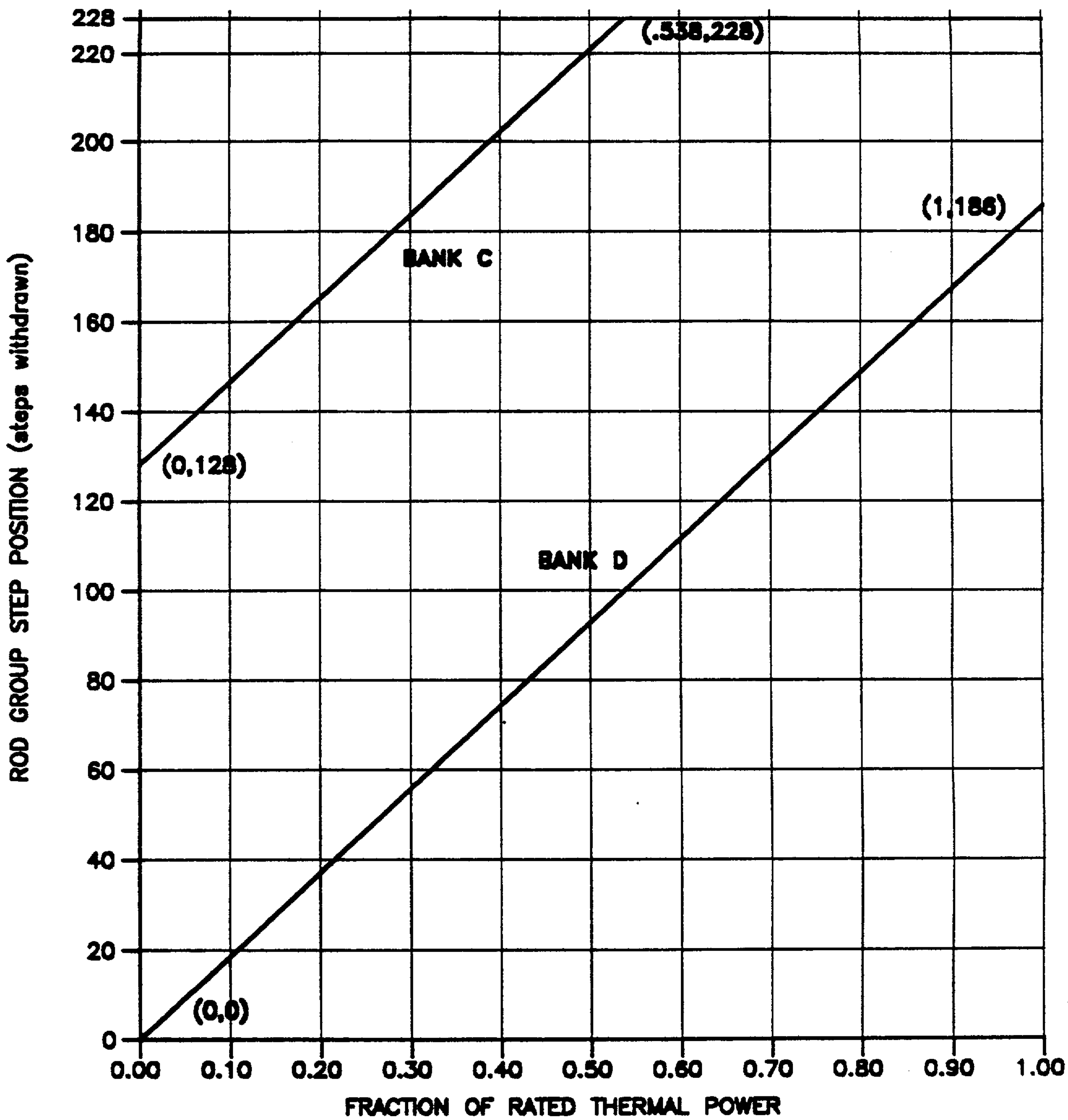


FIGURE 3.1-2
ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER
THREE-LOOP OPERATION

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 defined by Figure 3.2-1 for Relaxed Axial Offset Control (RAOC) operation, or

b. within a ± 3 percent target band about the target AFD 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 Figure 3.2-1 limits, either:

 1. Restore the indicated AFD to within the Figure 3.2-1 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 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 applicable target 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 Figure 3.2-1 limits.

*See Special Test Exception 3.10.2

**APL^{ND} is the minimum allowable power level for Base Load operation and will be provided in the Peaking Factor Limit Report per Specification 6.9.1.6.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

   2. At least once per hour for the first 24 hours after restoring the AFD Monitoring Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.3 When in Base Load operation, the target AFD of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 When in Base Load operation, the target AFD shall be updated at least once per 31 Effective Full Power Days by either determining the target AFD in conjunction with the surveillance requirements of Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

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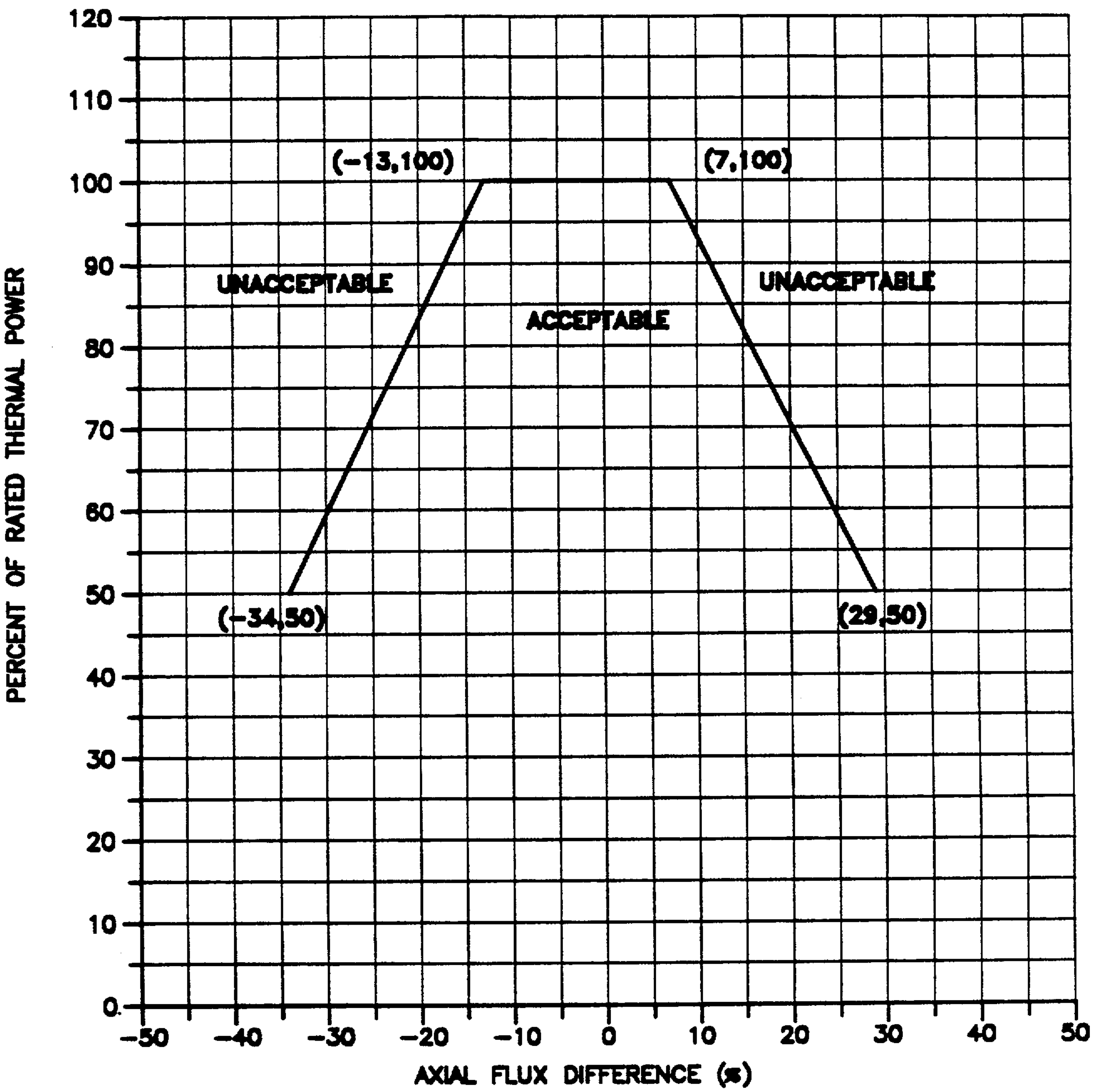

FIGURE 3.2-1
AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER FOR RAOC

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:

$$F_Q(Z) \leq \frac{2.32}{P} [K(Z)] \text{ FOR } P > 0.5$$

$$F_Q(Z) \leq (4.64) [K(Z)] \text{ FOR } P \leq 0.5$$

Where:

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

$K(Z)$ = the function obtained from Figure 3.2-2 for a given core height location.

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.

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 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:

$$F_Q^M(Z) \le \frac{2.32 \quad x\ K(Z)}{P\ x\ W(Z)} \text{ for } P > 0.5$$

$$F_Q^M(Z) \le \frac{2.32 \quad x\ K(Z)}{W(Z)\ x\ 0.5} \text{ for } P \le 0.5$$

where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.32 is the F_Q limit, K(Z) is given in Figure 3.2-2, 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. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.6.

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.

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*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

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

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

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

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:

$$\left\{ \left(\text{maximum} \quad \left[\frac{F_Q^M(Z) \times W(Z)}{\frac{2.32}{P} \times K(Z)} \right] \right) -1 \right\} \times 100 \text{ for } P \geq 0.5$$

$$\left\{ \left(\text{maximum} \quad \left[\frac{F_Q^M(Z) \times W(Z)}{\frac{2.32}{0.5} \times K(Z)} \right] \right) -1 \right\} \times 100 \text{ for } P < 0.5$$

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 of 3.2-1 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.

SURVEILLANCE REQUIREMENTS (Continued)

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 $\pm$ 3% of 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 FQ surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

$$APL^{BL} = \text{minimum} \left[\frac{2.32 \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}}\right] \times 100\%$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is 2.32. K(Z) is given in Figure 3.2-2. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during Base Load operation. The function is given in the Peaking Factor Limit Report as per Specification 6.9.1.6.

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.

SURVEILLANCE REQUIREMENTS (Continued)

c. Satisfying the following relationship:

$$F_Q^M(Z) \le \frac{2.32 \times K(Z)}{P \times W(Z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. The F_Q limit is 2.32.

K(Z) is given in Figure 3.2-2. P is the fraction of RATED THERMAL POWER. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.6.

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

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

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

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

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

POWER DISTRIBUTION LIMITS

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:

$$[(\text{max. of } [\frac{F_Q^M(Z) \times W(Z)_{BL}}{\frac{2.32}{P} \times K(Z)}]) - 1] \times 100 \text{ for } P \geq APL^{ND}$$

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.

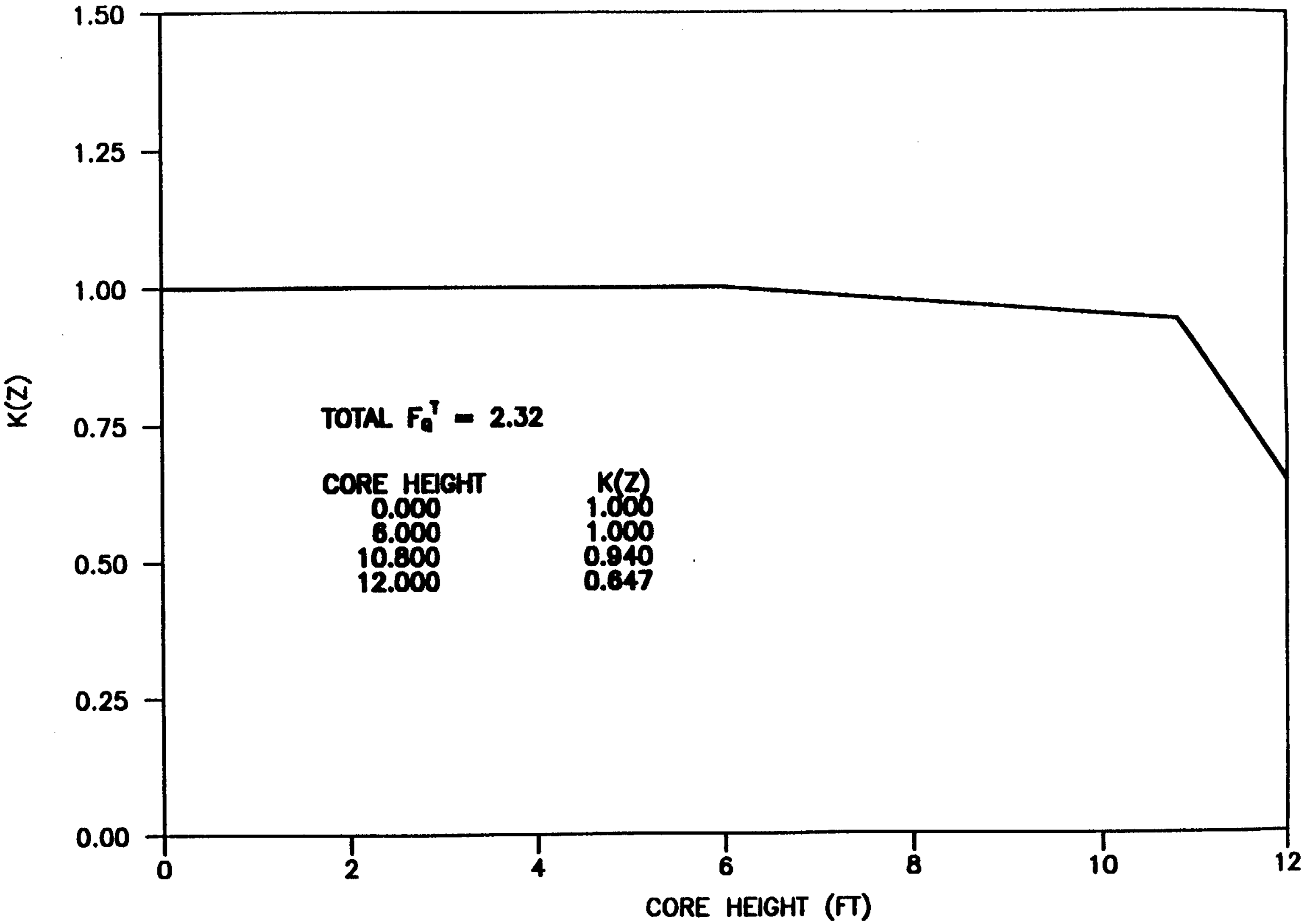

FIGURE 3.2-2
K(Z) - LOCAL AXIAL PENALTY FUNCTION FOR F_Q (Z)

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

LIMITING CONDITION FOR OPERATION

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

a. Measured RCS flow rate $\geq$ 292,800 gpm x $(1.0 + C_1)$, and

b. Measured $F^N_{\Delta H} \leq 1.49\ [1.0 + 0.3(1.0-P)]$

Where:

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

$F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map, and the measured values of $F^N_{\Delta H}$ shall be used for comparison above since the 1.49 value above accounts for a 4% allowance on incore measurement of $F^N_{\Delta H}$.

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

APPLICABILITY: MODE 1.

ACTION:

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

a. Within 2 hours either:

1. Restore RCS total flow rate and $F^N_{\Delta H}$ 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.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(12)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), #(3, 4), ##(4, 6), R(4, 5)	Q(15)	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q(15)	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q(15)	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	S	R(4, 5)	S/U(1), Q(8, 15)	N.A.	N.A.	2**, 3, 4, 5
7. Overtemperature ΔT	S	R(11)	Q(15)	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	Q(15)	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	Q(15)	N.A.	N.A.	1 (16)
10. Pressurizer Pressure--High	S	R	Q(15)	N.A.	N.A.	1, 2

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

**Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

***Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

#Each 31 Effective Full Power Days.

##Each 92 Effective Full Power Days.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, and evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) Quarterly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(9) Setpoint verification is not applicable.

(10) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the reactor trip breakers.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(11) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

(12) Verify that appropriate signals reach the undervoltage and shunt trip relays, for both the main and bypass breakers, from the manual reactor trip switch.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. In MODES 1 and 2 the most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1770 pcm is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. In MODES 3, 4, and 5, the most restrictive condition occurs at BOL, when the boron concentration is the greatest. In these modes, the required SHUTDOWN MARGIN is composed of a constant requirement and a variable requirement, which is a function of the RCS boron concentration. The constant SHUTDOWN MARGIN requirement is based on an uncontrolled RCS cooldown from a steamline break accident, as is the case for MODES 1 and 2. The variable SHUTDOWN MARGIN requirement is based on the results of boron dilution accident analyses, where the SHUTDOWN MARGIN is varied as a function of RCS boron concentration, to guarantee a minimum of 15 minutes for operator action prior to a loss of SHUTDOWN MARGIN.

Figure 3.1-1 must be used with a curve giving the required shutdown boron concentrations for various temperatures as a function of core burnup. This cycle dependent relationship is provided for each cycle in the plant Curve Book. From the Curve Book, a required boron concentration that will provide adequate SHUTDOWN MARGIN can be determined and this concentration may be used to enter Figure 3.1-1 to determine the specific required SHUTDOWN MARGIN for that condition.

The boron dilution analysis assumed a common RCS volume and dilution flow rate for MODES 3 and 4, which differed from the volume and flow rate assumed for MODE 5 analysis. The MODE 5 conditions assumed limited mixing in the RCS and cooling with the RHR system only. In MODES 3 and 4, it was assumed that at least one reactor coolant pump was operating. If at least one reactor coolant pump is not operating in MODE 4, then the SHUTDOWN MARGIN requirements for MODE 5 shall apply, provided that the dilution flow rate assumed in the MODE 5 Boron Dilution analysis is not exceeded.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; i.e., the positive limit is based on core conditions for all rods withdrawn, BOL, hot zero THERMAL POWER, and the negative limit is based on core conditions for all rods withdrawn, EOL, RATED THERMAL POWER. Accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

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 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 MTC value -42 pcm/°F. The MTC value of -33 pcm/°F 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 MTC value of -42 pcm/°F.

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 it 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 from full power equilibrium xenon conditions and requires 21,400 gallons of 7000 ppm borated water be maintained in the boric acid storage tanks or 436,000 gallons of 2000-2200 ppm borated water be maintained in the refueling water storage tank (RWST).

With the RCS temperature below 350°F, one boron injection flow path is acceptable without single failure consideration on the basis of the stable reactivity

BORATION SYSTEMS (Continued)

condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single boron injection flow path becomes inoperable.

The limitation for a maximum of one charging/safety injection pump (CSIP) to be OPERABLE and the Surveillance Requirement to verify all CSIPs except the required OPERABLE pump to be inoperable below 335°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide the required SHUTDOWN MARGIN as defined by Specification 3/4.1.1.2 after xenon decay and cooldown from 200°F to 140°F. This condition requires either 7100 gallons of 7000 ppm borated water be maintained in the boric acid storage tanks or 106,000 gallons of 2000-2200 ppm borated water be maintained in the RWST.

The gallons given above are the amounts that need to be maintained in the tank in the various circumstances. To get the specified value, each value had added to it an allowance for the unusable volume of water in the tank, allowances for other identified needs, and an allowance for possible instrument error. In addition, for human factors purposes, the percent indicated levels were then raised to either the next whole percent or the next even percent and the gallon figures rounded off. This makes the LCO values conservative to the analyzed values. The specified percent level and gallons differ by less than 0.3%.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The BAT minimum temperature of 65°F ensures that boron solubility is maintained for concentrations of at least the 7750 ppm limit. The RWST minimum temperature is consistent with the STS value and is based upon other considerations since solubility is not an issue at the specified concentration levels. The RWST high temperature was selected to be consistent with analytical assumptions for containment heat load.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

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 1.30 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;

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

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.

POWER DISTRIBUTION LIMITS

BASES

AXIAL FLUX DIFFERENCE (Continued)

At power levels below APL^{ND}, the limits on AFD are defined by Figure 3.2-1, i.e., that defined by the 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 limits of which are defined by Figure 3.2-1, and 2) Base Load operation, which is defined as the maintenance of the AFD within a ± 3% 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 value, the Base Load operating procedure restricts the indicated AFD to a relatively small target band and power swings (AFD target band of ± 3%, $APL^{ND} \leq$ 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 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.

AXIAL FLUX DIFFERENCE (Continued)

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 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 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;

FIGURE B 3/4 2-1 DELETED

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.

$F^N_{\Delta H}$ 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 $F^N_{\Delta H}$ ensures that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

$F^N_{\Delta H}$ is evaluated as being less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR completely offset any rod bow penalties. The applicable value of rod bow and any other penalties is presented in FSAR Section 4.4.2.2.4.2. This margin includes the following:

a. Design limit DNBR of 1.30 vs 1.28,
b. Grid Spacing (K_s) of 0.046 vs 0.059,
c. Thermal Diffusion Coefficient of 0.038 vs 0.059,
d. DNBR Multiplier of 0.86 vs 0.88, and
e. Pitch reduction.

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 $F^M_Q(Z)$ 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) function for normal operation is provided in the Peaking Factor Limit Report per Specification 6.9.1.6.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 45.0 psig and a peak air temperature of 380°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly normally containing 264 fuel rods clad with Zircaloy-4 except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, stainless steel, or by vacancies may be made in fuel assemblies if justified by a cycle-specific evaluation. Should more than a total of 30 fuel rods or more than 10 fuel rods in any one assembly be replaced per refueling a Special Report describing the number of rods replaced will be submitted to the Commission, pursuant to Specification 6.9.2, within 30 days after cycle startup. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.2 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 52 shutdown and control rod assemblies. The shutdown and rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium, or 95% hafnium with the remainder zirconium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 9410 ± 100 cubic feet at a nominal T_{avg} of 588.8°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological station shall be located as shown on Figure 5.1-1.

PEAKING FACTOR LIMIT REPORT

6.9.1.6 The W(Z) Functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be established for each reload core and implemented prior to use.

The methodology used to generate the W(Z) functions for RAOC and Base Load operation and the value for APL^{ND} shall be those previously reviewed and approved by the NRC.* If changes to these methods are deemed necessary, they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

A report containing the W(Z) function for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be provided to the NRC in accordance with 10 CFR 50.4 within 30 days after each cycle initial criticality.

Any information needed to support W(Z), $W(Z)_{BL}$, and APL^{ND} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC 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;

* WCAP-10216, "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification."

PEAKING FACTOR LIMIT REPORT (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. 7 TO FACILITY OPERATING LICENSE NO. NPF-63

CAROLINA POWER & LIGHT COMPANY, et al.

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By letters dated February 1 and February 8, 1988, the Carolina Power & Light Company, the licensee for the Shearon Harris Nuclear Power Plant, Unit 1 (Shearon Harris Unit 1), submitted requests for changes to the Technical Specifications (TS) and safety evaluations to support operation of Cycle 2 (References 1 and 2). The proposed changes refer to control rod Bank-D configuration, increases in the radial and total peaking factors $F_{\Delta H}$ and F_Q, the introduction of a boron dilution/sliding shutdown margin and some miscellaneous technical specifications. The proposed changes are needed so that the licensee can use higher enrichment fuel (up to 4.2 weight percent uranium 235), extend the fuel irradiation limits, and permit operation of longer fuel cycles. The environmental consequences of the use of higher enrichment fuel and extended irradiation were addressed in licensee submittals associated with Amendment 5 dated May 26, and November 2, 1987.

2.0 EVALUATION

2.1 D-Bank Reconfiguration

Currently, control rod Bank-D consists of four rod cluster control assemblies located close to the core periphery. This configuration (which differs from the standard Westinghouse design) was provided to enhance load following operation during the first cycle. However, Cycle 2 operation will be based on a low leakage configuration, with new assemblies located in the interior of the core. Thus, the present Bank-D configuration will have reduced worth and when inserted will increase the new assembly power peaking. In addition, the licensee intends to operate Shearon Harris Unit 1 as a base load plant during Cycle 2. Therefore, the four peripheral control rod assemblies are inappropriate. The proposed configuration consists of eight control rod assemblies uniformly distributed throughout the core. The proposed control rod Bank-D configuration is identical with the generic Westinghouse 3-loop 17x17 arrangement of fuel assemblies and control rods in the core.

An evaluation was performed to determine the effect of the proposed configuration on the FSAR accident analyses in conjunction with the Relaxed Axial Offset Control (RAOC) and $F_{\Delta H}$ proposed changes which are discussed below. Results of analyses performed by the licensee indicate that the proposed insertion limits of the new control rod Bank-D configuration will maintain power distribution limits, shutdown margin, reactor trip reactivity, ejected rod worth and associated peaking factors, dropped rod worths, and differential and integral rod worths within the values assumed in the FSAR non-LOCA analyses. Thus, the FSAR analyses remain valid. For the small break LOCA analysis, reactor trip and rod insertion of all banks is assumed; therefore, the proposed control rod Bank-D configuration has no effect. Finally, for the large break LOCA, no credit is taken for control rod insertion; thus, the proposed reconfiguration has no effect.

In summary, we find that the proposed control rod Bank-D reconfiguration is necessary for the intended operation of Cycle 2 in Shearon Harris; and the effects of the proposed insertion limits are within the FSAR analyses limits and are acceptable. Therefore, the proposed TS change in Fig. 3.1-1 is acceptable.

2.2 F_Q INCREASE, RAOC/BASE LOAD OPERATION AND F_Q SURVEILLANCE

2.2.1 Discussion

There are two fundamental changes proposed: (1) increase of the heat flux hot channel factor F_Q from 2.28 to 2.32 and (2) replacement of the Constant Axial Offset Control (CAOC) procedure with RAOC Baseload operating strategy and direct F_Q surveillance. The implementation of these changes involves the following TS: (a) 3.2.2 to increase F_Q,(b) Fig. 3.2-2 to revise the Local Axial Penalty Function K(z), (c) 3/4.2.1 to replace the existing Constant Axial Offset Control with the RAOC Baseload operation, (d) 4.2.2.1 to replace the $F_{xy}(z)$ with the F_Q surveillance, (e) Table 2.2-1 to revise the f(ΔI) reset function, and (f) 6.9.1.6 to define the requirements of the radial peaking factor report.

The basis for the proposed increase in F_Q from 2.28 to 2.32 is the LOCA reanalysis with the BASH Rev. 2 code (Ref. 3). This code allows a more realistic thermal/hydraulic core simulation. The result of this analysis allows the proposed increased F_Q because the calculated peak clad temperature, maximum local metal/water reaction and total core metal/water reaction are within the ECCS acceptance criteria of 10 CFR 50.46. The use of the BASH Rev. 2 code has been accepted by the staff for large LOCA evaluations; therefore, this change is acceptable. The other proposed change is a combination of RAOC Baseload operating procedures with direct F_Q surveillance to enhance operational flexibility. It is proposed to replace the existing CAOC procedures with the RAOC Baseload, as described in Reference 4, which has received staff approval. In the event that high peaking factors would preclude operation at 100% of rated power, base load operation will be restricted to a narrow CAOC band of ±3%

of a target value above a predetermined power level. This power level will be 85% for Cycle 2 and be justified in the surveillance report as required by TS 6.9.1.6. The proposed RAOC procedures are associated with a direct F_Q surveillance (Ref. 4). Currently, TS 4.2.2.2 requires periodic plant surveillance on $F_{xy}(z)$ as partial verification for $F_Q(z)$. The proposed TS 3/4.2.1 provides for direct monitoring of $F_Q(z)$. The concept and the method of direct F_Q surveillance have been accepted by the staff and are described in Ref. 4. Finally, the provisions of the radial peaking factor report required by TS 6.9.1.6 are modified to reflect the changes to F_Q as described above.

There are two operational modes covered in the proposed TS, (a) the RAOC operation for which the axial flux differences are calculated to create F_Qs which satisfy the power peaking factor limits and (b) the Base Load operation (for power levels above 85%) for which the axial flux difference is limited within a band of ±3% of a target value. Surveillance requirements of TS 4.2.2.2 account for both modes of operation. This is accomplished in terms of a multiplier function W(z) that modifies the routine flux mapping fluxes which are then compared to the TS limits described above. The W(z) function is also provided in the report required by TS 6.9.1.6. Finally, accident flux shapes were generated at the core limits, which were analyzed to estimate the clad yield stresses. It was determined that the negative wing of f(ΔI) should have an intercept at -25% and a slope of 2.36%/%. This is more restrictive than the RAOC limit and is also reflected in the TS change. Therefore, the proposed TS change fulfills the intent of the original TS, i.e., protecting against specified power peaking limits and direct measurement surveillance of the local flux and power.

The new axial flux difference limits form the basis for new axial distributions assumed in the non-LOCA safety analysis. This power envelope, defines the axial offset terms for the overtemperature-ΔT and overpressure-ΔT (OTΔT/OPΔT) trip functions. For this new power distribution, it must be shown that DNB is within the design basis. For power distributions at nominal full power conditions for analyses which do not rely on the OTΔT/OPΔT trip function, the licensee established that the FSAR analyses continue to be valid for operation with RAOC. Likewise, the existing OTΔT/OPΔT setpoints were compared to the revised core limits, including the effects of RAOC and the increased $F_{\Delta H}$, and are not impacted by these changes. For core power less than full power and the additional power shapes allowed by RAOC, the revised f(ΔI) function is necessary to protect against DNB. The positive wing of f(ΔI) has an intercept at 7.0% and slope of 1.7%/% (This is in addition to the negative limit discussed above). The change to RAOC does not affect the existing OTΔT/OPΔT setpoints and, therefore, the conclusions of the non-LOCA safety analyses presented in the FSAR remain valid.

For the LOCA analysis, the limiting power shapes for a large and a small break LOCA were estimated and justified to be bounding within the total axial offset band. This offset for cycle 2 at full power is -20% to +13%. However, by the addition of the RAOC/Baseload operation, the axial offset required by the core is bounded by the FSAR existing limits; therefore, the existing ECCS analysis is not affected.

2.2.2 Technical Specification Changes

To implement the F_Q increase, the RAOC/Base Load operation and the F_Q surveillance the following TS must be changed:

(a) TS 3.2.2, increasing the heat flux hot channel factor;

(b) TS Fig.3.2.2, revising the local axial penalty function K(z);

(c) TS 3/4.2.1, replacing CAOC with RAOC/Base Load operation;

(d) TS 4.2.2.1 surveillance, replacing F_{xy} with F_Q;

(e) TS Table 2.2-1, revising the f(ΔI) function; and

(f) TS 6.9.1.6, delineating the contents of the revised peaking factor limit report.

The discussion in Section 2.2.1 indicates that the proposed changes to F_Q and the associated axial function K(z) do not affect the LOCA and non-LOCA analyses of the FSAR. Thus, the consequences of previously evaluated accidents are not increased as a result of these changes. Replacing the CAOC with RAOC/Base Load operation and revising the f(ΔI) function maintains the validity of the analyses in the FSAR. The replacement of the F_{xy} with F_Q surveillance maintains the intent of the technical specification for a direct measurement of the flux and power distribution. The change in the TS 6.9.1.6 is administrative in nature. Therefore, the proposed changes are acceptable.

2.3 Increase of $F_{\Delta H}$

2.3.1 Discussion

The licensee proposed to increase the radial peaking factor equation from: $F_{\Delta H} \leq 1.49x[1+0.2(1-P)]$

to: $F_{\Delta H} \leq 1.49x[1+0.3(1-P)]$

i.e., the existing multiplier has been increased to 0.3. This change allows higher values for $F_{\Delta H}$ at reduced power level. The licensee's motivation for this change is for greater operational flexibility at reduced power levels. The criterion for the acceptability of this change is the value of the minimum DNBR to be no less than ≥ 1.30. The core limit curves in TS Fig. 2.1-1 represent thermal power, RCS pressure and average temperature for core states for which minimum departure from nucleate boiling ratio (MDNBR) ≥ 1.30. The full power value of $F_{\Delta H}$ remains the same. Therefore, the LOCA analyses presented in the FSAR, which assume 100% power operation, are not affected. The proposed change does affect event analyses at a fractional power level and events which rely on OTΔT/OPΔT trip function for protection. The existing setpoints were compared to the revised core limits with the 0.3 $F_{\Delta H}$ multiplier, and it was determined that the existing setpoints envelope the new limits. Therefore, the non-LOCA events analyses presented in the FSAR are still valid.

2.3.2 Technical Specification Changes

The implementation of the increased $F_{\Delta H}$ values for fractional core power levels require changes in:

(a) TS 3.2.3.6, the equation used to determine $F_{\Delta H}$; and

(b) TS Figure 2.1-1, core limit curves.

The discussion in Section 2.3.1 indicates that the proposed change is enveloped by the existing LOCA and non-LOCA analyses in the FSAR and, therefore, are acceptable.

2.4 Miscellaneous Technical Specification Changes

2.4.1 Discussion

It is proposed that TS Table 4.3-1 on the surveillance requirements for the excore power range monitors and TS 5.3.1 describing the fuel assemblies be modified as follows: (a) TS Table 4.3-1 currently requires a single point INCORE/EXCORE axial flux difference be performed monthly and an INCORE/EXCORE calibration be performed quarterly. It is proposed that the surveillance and calibration requirements be modified to 31 and 92 effective full power days respectively and (b) TS 5.3.1 for the assembly description is proposed to be modified to allow for repairs of damaged fuel rods by substitutions with filler rods or vacancies.

The purpose of the surveillance and calibration of the excore detectors for the axial flux difference is to account for changes in the core caused by fuel and burnable poison depletion. Time by itself plays no role in these changes because the detectors are stable and last for a very long time. Therefore, it is reasonable to modify the calendar monthly surveillance to 31 effective full power days and the calendar quarterly calibration to 92 effective full power days. In addition, other TS provide for quarterly calibration should extended operation at low power levels or periods of extended shutdown reach these limits.

The assembly description requires 264 Zircaloy-4 clad fuel rods. The modification provides for the substitution of a limited number of fuel rods by stainless steel, Zircaloy or vacancies if they are justified by a cycle specific evaluation. This evaluation will assure that all aspects, i.e., physics and thermal hydraulics, will meet the requirements of the intact assembly.

2.4.2 Evaluation of Technical Specification Changes

The proposed change in the wording INCORE/EXCORE flux difference is equivalent to the current specification with respect to the anticipated changes in the detectors, the core and the associated electronics; and, thus, it is acceptable. The proposed changes in TS 5.3.1 on the fuel assembly description include the requirement that whatever substitutions in fuel rods are made they must be justified by cycle specific analyses. These analyses are to be performed with NRC approved codes and comply with the requirements of an intact assembly. In addition, a maximum number of assembly and core substitutions are specified. Therefore, such changes are acceptable.

2.5 Boron Dilution/Sliding Shutdown Margin

2.5.1 Discussion

The proposed amendment modifies TS 3/4.1., Shutdown Margin, from a fixed to a variable shutdown margin requirement. Due to increased enrichment and low leakage loading configuration, Shearon Harris will have higher shutdown margin requirements at the beginning of Cycle 2 (BOC-2) than the corresponding value at BOC-1 (Ref. 2).

The shutdown margin ensures that (a) the reactor can be made subcritical from any operating condition, (b) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (c) the reactor will be maintained subcritical to preclude inadvertent criticality in the shutdown mode. The value of the shutdown margin requirement is determined through postulated reactivity transients. The limiting transients are: main steamline break and inadvertent boron dilution depending on the mode of operation and time in the cycle for modes 3, 4 and 5. Beginning of cycle (BOC) is the most limiting time period (when boron concentration is at its maximum). In these modes, the required shutdown margin consists of a constant and a variable component. The constant part arises from the requirements of a steam line break reactor coolant system cooldown. The variable part arises from the requirements of an inadvertent boron dilution event, where the shutdown margin is varied as a function of primary boron concentration, to guarantee a minimum of 15 minutes for operator action prior to loss of shutdown margin. The existing TS require the shutdown margin to be fixed at the higher BOC value throughout the cycle, even though the required shutdown margin decreases with core burnup. The proposed change alleviates this condition by introducing a variable shutdown margin for operating modes 3, 4 and 5 to take credit for reduced shutdown margin toward the end of cycle. Rod insertion limits (in TS 3.1.3.6) ensure adequate negative reactivity to take the reactor subcritical from modes 1 and 2. Therefore, the variable shutdown margin is not applicable in these modes.

2.5.2 Technical Specification Changes

The implementation of the proposed boron dilution sliding/shutdown margin change affects the following specifications:

(a) TS 3/4.1.1.1 and 3/4.1.1.2, Boration Control, Flow Path, Shutdown

(b) TS 3/4.1.2.5 and 3/4.1.2.6, Boration Control, Borated Water Source, Shutdown.

The proposed reduction in shutdown margin impacts only the inadvertent boron dilution, which has been reanalyzed. It was determined that the proposed sliding shutdown margins maintain at least 15 minutes from alarm indication to loss of shutdown margin, as required by section 15.4.6 of the Standard Review Plan. The shutdown margin requirements for modes 1, 2, 3 and 4 remain equal to or greater than those in the existing TS, and the main steam line break event is not affected by the revision of the shutdown margin requirements. Therefore, the proposed revisions to TS 3/4.1.1.1 and 3/4.1.1.2 do not involve a reduction in the margin of safety.

The minimum borated water volumes provided in TS 3/4.1.2.5 and 3/4.1.2.6 were increased to assure that the required shutdown margins can be provided. Therefore, the proposed revisions do not involve a reduction in the margin of safety; thus, they are acceptable.

2.6 ROD BOW PENALTY

The proposed change is in the bases of TS 3/4.2.2 and 3/4.2.3 and consists of the replacement of a parenthetical note on rod bow with a reference to the FSAR which contains this and other rod bow penalties. This change is adminstrative in nature and does not effect the specification; therefore, it is acceptable.

2.7 DESIGN BASIS ACCIDENT ANALYSIS RELATIVE TO EXTEND FUEL BURNUP

The licensee has requested authorization to increase fuel enrichment to 4.2 weight percent of U-235 and to allow fuel burnup up to 60,000 megawatt days per metric ton (MWD/MT). The staff and licensee evaluated the potential impact of this change on the radiological assessment of design basis accidents (DBA) which were previously analyzed in the licensing of the Shearon Harris Unit 1 nuclear power plant.

The licensee, in their submittals of May 26 and November 2, 1987, concluded that the design basis accidents previously analyzed by the licensee in their FSAR bound any potential radiological consequences of DBA that could result with the extended fuel burnup fuel.

The staff reviewed the licensee's submittals and also reviewed a publication which was prepared for the NRC entitled, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors," NUREG/CR 5009, February 1988. The NRC contractor, the Pacific Northwest Laboratory (PNL) of Battelle Memorial Institute, examined the changes that could result in the NRC DBA assumptions, described in the various appropriate SRP sections and/or Regulatory Guides, that could result from the use of extended burnup fuel (up to 60,000 MWD/MT). The staff agrees that the only DBA that could be affected by the use of extended burnup fuel, even in a minor way, would be the potential thyroid doses that could result from a fuel handling accident. PNL estimates that I-131 fuel gap activity in the peak fuel rod with 60,000 MWD/MT burnup could be as high as 12%. This value is approximately 20% higher than the value normally used by the staff in evaluating fuel handling accidents (Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facilities for Boiling and Pressurized Water Reactors).

The staff, therefore, reevaluated the fuel handling accidents for the Shearon Harris Unit 1 facility with an increase in iodine gap activity in the fuel damaged in a fuel handling accident. Table 1 presents the fuel handling accident thyroid doses presented in the operating licensing Safety Evaluation Report, dated November 1983, and the increased thyroid doses (by 20%) resulting from extended burnup fuel.

Table 1

Thyroid Doses as a Consequence of DBA Fuel Handling Accidents

	Exclusion Area		Low Population Zone	
	Thyroid Dose (Rem)		Thyroid Dose (Rem)	
Fuel Handling Accident	A*	B**	A*	B**
In Fuel Building	5.5	6.6	1.5	1.8
In Reactor Building	5.0	6.0	5.0	6.0

*A SER dose
**B Extended fuel burnup dose

The staff concludes that the only potential increased doses potentially resulting from DBA with extended fuel burnup to 60,000 MWD/MT is the thyroid dose resulting from fuel handling accidents and these doses remain well within the 300 Rem thyroid exposure guideline values set forth in 10 CFR Part 100 and that this small calculated increase is not significant.

3.0 SUMMARY

The staff has reviewed the information submitted by Carolina Power & Light Company, the licensee for the Shearon Harris Unit 1 plant, to support proposed Technical Specification changes required for the operation of Cycle 2. The proposed amendment affects the configuration of control rod Bank-D, requested increases in the values of the radial and total peaking factors, proposed a boron dilution sliding/shutdown margin, requested changes in the INCORE/EXCORE surveillance and calibration intervals, requested changes in the description of the fuel assembly, and finally requested a change in the bases of the rod bow penalty.

Our evaluation indicates that the requested amendments are acceptable. The change in TS 5.3.1 regarding the rod bow penalty is an administrative change in the reference and affects the bases but not the Technical Specification and, therefore, is acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register (53 FR 30355) on August 11, 1988. Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the Federal Register (53 FR 17777) on May 18, 1988, 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Lambros Lois
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Dated: August 16, 1988

REFERENCES

1. Letter (NLS-88-023) from L. W. Eury, Carolina Power and Light Co., to USNRC, "Request for License Amendment Cycle 2 Operation," dated February 1, 1988.

2. Letter (NLS-88-031) from L. W. Eury, Carolina Power and Light Co., to USNRC, "Request for License Amendment Boron Dilution/Sliding Shutdown Margin," dated February 8, 1988.

3. WCAP-10266PA, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," by J. N. Kabadi et al., Westinghouse, dated August 1986.

4. WCAP-10216-PA, "Relaxation of Constant Axial Offset Control, F_Q Surveillance Technical Specification" by R. W. Miller et al., dated June 1983.

AMENDMENT NO. 7 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

Docket File
NRC PDR
Local PDR
PDII-1 Reading
S. Varga (14E4)
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E. Adensam
P. Anderson
B. Buckley
OGC
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
J. Partlow (9A2)
T. Barnhart (4) (P1-137)
W. Jones (P-130A)
E. Butcher (11F23)
L. Lois
I. Spickler
ACRS (10)
GPA/PA
ARM/LFMB

cc: Licensee/Applicant Service List

OF01
1/1