



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

DUKE POWER COMPANY

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39  
License No. NPF-35

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-35 filed by the Duke Power Company acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc., (licensees) dated November 13, 1987 and supplemented December 11, 1987, and January 15 and 20, 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 as set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, as amended, 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 set forth in 10 CFR Chapter I;
  - 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 hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-35 is hereby amended to read as follows:

8803030277 880216  
PDR ADOCK 05000413  
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 39, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

*Kahtan N. Jabbour*

Kahtan N. Jabbour, Acting Director  
Project Directorate II-3  
Division of Reactor Projects I/II

Attachment:  
Technical Specification Changes

Date of Issuance: February 16, 1988

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 31, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and changes to Technical Specifications concerning the Boron Dilution Mitigation System are effective when the system is installed and declared operable.

FOR THE NUCLEAR REGULATORY COMMISSION

Kahtan N. Jabbour, Acting Director  
Project Directorate II-3  
Division of Reactor Projects I/II

Attachment:  
Technical Specification Changes

Date of Issuance: February 16, 1988

\* SEE PREVIOUS CONCURRENCE

PDII-3/DRPI/II  
MR00d/mac  
02/4/88

*KNS*  
PDII-3/DRPI/II  
KJabbour  
02/10/88

OGC-WF  
\*MYoung  
01/29/88

*KNS*  
PDII-3/DRPI/II  
Acting PD  
02/10/88



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

DUKE POWER COMPANY

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31  
License No. NPF-52

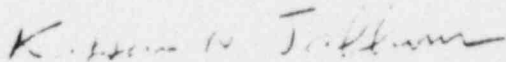
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-52 filed by the Duke Power Company acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency, (licensees) dated November 13, 1987, and supplemented December 11, 1987, and January 15 and 20, 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 as set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, as amended, 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 set forth in 10 CFR Chapter I;
  - 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 hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-52 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 31, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and changes to Technical Specifications concerning the Boron Dilution Mitigation System are effective when the system is installed and declared operable.

FOR THE NUCLEAR REGULATORY COMMISSION



Kahtan N. Jabbour, Acting Director  
Project Directorate II-3  
Division of Reactor Projects I/II

Attachment:  
Technical Specification Changes

Date of Issuance: February 16, 1988

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 39, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kahtan N. Jabbour, Acting Director  
Project Directorate II-3  
Division of Reactor Projects I/II

Attachment:  
Technical Specification Changes

Date of Issuance: February 16, 1988

PDII-3/DRPI/II  
MK000/mac  
02/4/88

*KNJ*  
PDII-3/DRPI/II  
KJabbour  
01/15/88

OGC-WF  
MYoung\*  
01/ /88

*if JS*  
PDII-3/DRPI/II  
Acting PD  
01/10/88



ATTACHMENT TO LICENSE AMENDMENT NO. 39

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO. 31

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf page is also provided to maintain document completeness.

| <u>Amended<br/>Page</u> | <u>Overleaf<br/>Page</u> |
|-------------------------|--------------------------|
| 2-8                     |                          |
| 3/4 1-2                 |                          |
| 3/4 1-3                 |                          |
| 3/4 1-21                |                          |
| 3/4 1-22                |                          |
| 3/4 1-23 (deleted)      |                          |
| 3/4 2-1                 |                          |
| 3/4 2-2                 |                          |
| 3/4 2-3                 |                          |
| 3/4 2-4 (deleted)       |                          |
| 3/4 2-5                 |                          |
| 3/4 2-6                 |                          |
| 3/4 2-7                 |                          |
| 3/4 2-7a                |                          |
| 3/4 2-7b                |                          |
| 3/4 2-7c                |                          |
| 3/4 2-7d (deleted)      |                          |
| 3/4 3-2                 |                          |
| 3/4 3-6                 |                          |
| 3/4 3-12                |                          |
| 3/4 3-55                | 3/4 3-56                 |
| 3/4 3-92a               |                          |
| 3/4 3-92b               |                          |
| 3/4 9-1                 |                          |
| 3/4 9-1a                |                          |
| 3/4 9-2 (deleted)       |                          |
| B 3/4 2-1               |                          |
| B 3/4 2-2               |                          |
| B 3/4 2-2a              |                          |
| B 3/4 2-3 (deleted)     |                          |
| B 3/4 2-4               |                          |
| B 3/4 2-4a (deleted)    |                          |
| B 3/4 9-1               |                          |
| 6-19                    |                          |
| 6-19a (deleted)         |                          |

TABLE 2.2-1 (Continued)  
TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

|       |        |   |
|-------|--------|---|
| $T'$  | $\leq$ | 590.8°F (Nominal $T_{avg}$ allowed by Safety Analysis); |
| $K_3$ | =      | 0.001189;   |
| $P$   | =      | Pressurizer pressure, psig;                             |
| $P'$  | =      | 2235 psig (Nominal RCS operating pressure);             |
| $S$   | =      | Laplace transform operator, $s^{-1}$ ;                  |

and  $f_1(\Delta I)$  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:

- (i) For  $q_t - q_b$  between -22.5% and -6.5%,  
 $f_1(\Delta I) = 0$ , 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;
- (ii) For each percent that the magnitude of  $q_t - q_b$  is more negative than -22.5%, the  $\Delta T$  Trip Setpoint shall be automatically reduced by 3.151% of its value at RATED THERMAL POWER; and
- (iii) For each percent that the magnitude of  $q_t - q_b$  is more positive than -6.5%, the  $\Delta T$  Trip Setpoint shall be automatically reduced by 1.641% 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 2.4%.



REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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- e. When in MODE 3 or 4, 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.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within  $\pm 1\% \Delta k/k$  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.

## REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN -  $T_{avg} \leq 200^{\circ}\text{F}$

### LIMITING CONDITION FOR OPERATION

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3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1%  $\Delta k/k$ .

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1%  $\Delta k/k$ , 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

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4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1%  $\Delta k/k$ :

- 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.

## REACTIVITY CONTROL SYSTEMS

### CONTROL BANK INSERTION LIMITS

#### LIMITING CONDITION FOR OPERATION

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3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-1.

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 the above figure, or
- c. Be in at least HOT STANDBY within 6 hours.

#### SURVEILLANCE REQUIREMENTS

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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.

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\*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

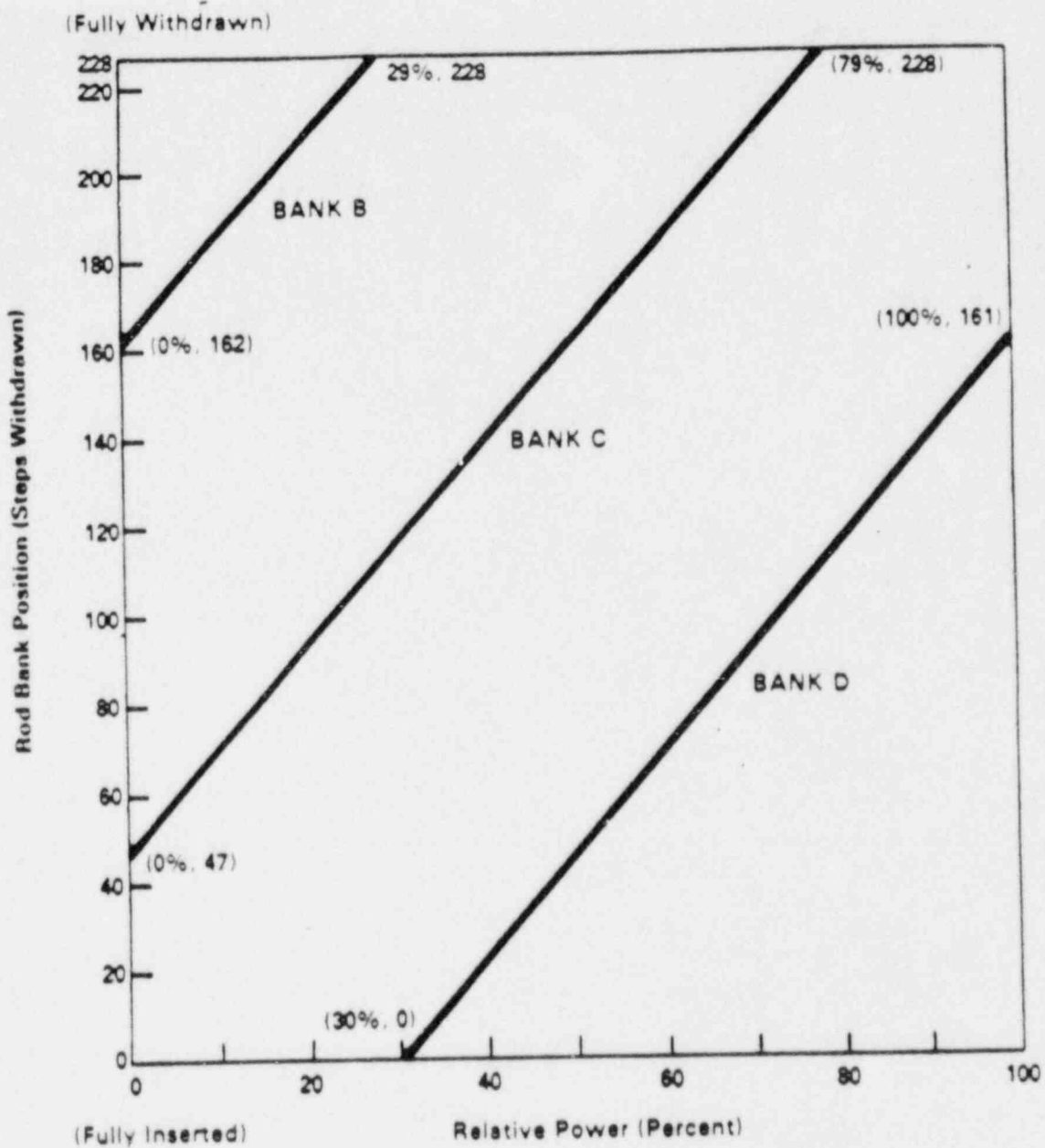


FIGURE 3.1-1  
 ROD BANK INSERTION LIMITS VERSUS THERMAL POWER  
 FOUR LOOP OPERATION

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### 3/4.2 POWER DISTRIBUTION LIMITS

#### 3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

##### LIMITING CONDITION FOR OPERATION

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

- a. the allowed operational space defined by Figure 3.2-1 for RAOC operation, or
- b. within a  $\pm 3\%$  target band about the target flux difference during baseload 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,
  1. Either 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<sup>ND\*\*</sup> with the indicated AXIAL FLUX DIFFERENCE outside of the applicable target band about the target flux difference:
  1. Either restore the indicated AFD to within the target band limits within 15 minutes, or
  2. Reduce THERMAL POWER to less than APL<sup>ND</sup> 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.

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\*See Special Test Exceptions Specification 3.10.2.

\*\*APL<sup>ND</sup> is the minimum allowable power level for base load operation and will be provided in the Peaking Factor Limit Report per Specification 6.9.1.9.



## POWER DISTRIBUTION LIMITS

### LIMITING CONDITION FOR OPERATION

#### 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 Monitor 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.
- c. The provisions of Specification 4.0.4 are not applicable.

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

4.2.1.3 When in Base Load operation, the target axial flux difference 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 flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference in conjunction with the surveillance requirements of Specification 3/4.2.2 or by linear interpolation between the most recently measured values and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

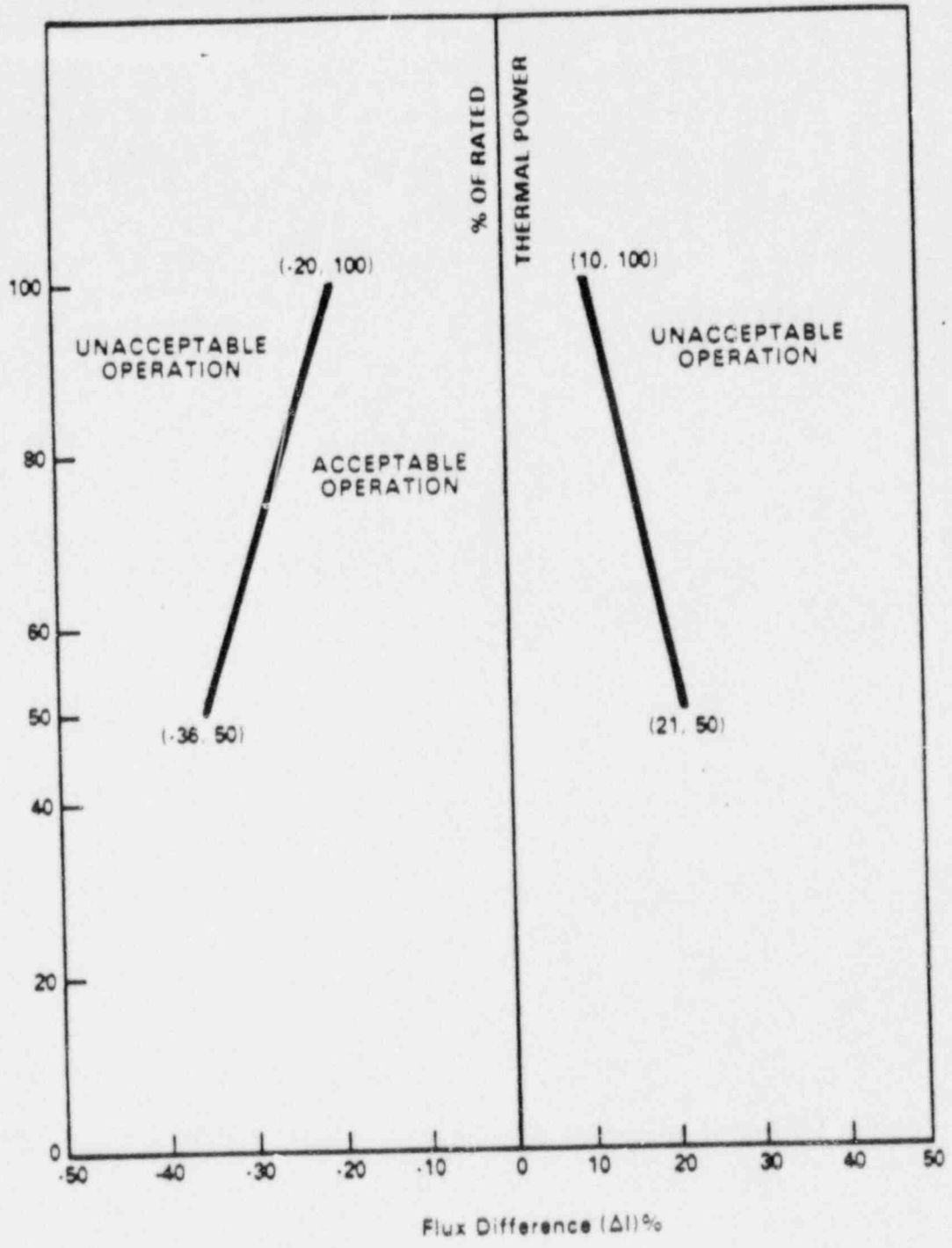


FIGURE 3.2-1  
 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

Pages 3/4 2-4 through 3/4 2-4c intentionally deleted.

POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION

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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:  $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$ , 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  $\Delta T$  Trip Setpoints (value of  $K_4$ ) have been reduced at least 1% (in  $\Delta T$  span) for each 1%  $F_Q(Z)$  exceeds the limit, and
- 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

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4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

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

$$F_Q^M(z) \leq \frac{2.32}{P \times W(z)} \times K(z) \quad \text{for } P > 0.5$$

$$F_Q^M(z) \leq \frac{2.32}{W(z) \times 0.5} \times K(z) \quad \text{for } P \leq 0.5$$

where  $F_Q^M(z)$  is the measured  $F_Q(z)$  increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.32 is the  $F_Q$  limit,  $K(z)$  is given in Figure 3.2-2,  $P$  is the relative 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 6.9.1.9.

- 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.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

e. With measurements indicating

$$\begin{array}{l} \text{maximum} \\ \text{over } z \end{array} \frac{F_Q^M(z)}{K(z)}$$

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

$$\begin{array}{l} \text{maximum} \\ \text{over } z \end{array} \frac{F_Q^M(z)}{K(z)} \text{ is not increasing.}$$

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:

$$\left\{ \left( \begin{array}{l} \text{maximum} \\ \text{over } z \end{array} \left[ \frac{F_Q^M(z) \times W(z)}{\frac{2.32}{P} \times K(z)} \right] - 1 \right) \times 100 \quad \text{for } P \geq 0.5 \right.$$
$$\left. \left\{ \left( \begin{array}{l} \text{maximum} \\ \text{over } z \end{array} \left[ \frac{F_Q^M(z) \times W(z)}{\frac{2.32}{0.5} \times K(z)} \right] - 1 \right) \times 100 \quad \text{for } P < 0.5 \right.$$

- 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.



## POWER DISTRIBUTION LIMITS

### 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:

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

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 Peak Factor Limit Report as per Specification 6.9.1.9.

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

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

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

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

c. Satisfying the following relationship:

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

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

$K(Z)$  is given in Figure 3.2-2.  $P$  is the relative THERMAL POWER.  $W(Z)_{BL}$  is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.9.

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 surveillance 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

$$\begin{array}{l} \text{maximum} \\ \text{over } z \end{array} \quad \left[ \frac{F_Q^M(z)}{K(z)} \right]$$

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 4.2.2.4c, or
2.  $F_Q^M(Z)$  shall be measured at least once per 7 EFPD until 2 successive maps indicate that

$$\begin{array}{l} \text{maximum} \\ \text{over } z \end{array} \quad \left[ \frac{F_Q^M(z)}{K(z)} \right] \text{ is not increasing.}$$

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

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

## POWER DISTRIBUTION LIMITS

### SURVEILLANCE REQUIREMENTS (Continued)

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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:

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

- g. The limits specified in 4.2.2.4c., 4.2.2.4e., and 4.2.2.4f. above are not applicable in the following core plan 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.

Pages 3/4 2-7d through 3/4 2-7f intentionally deleted.

TABLE 3.3-1

## REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT                                      | TOTAL NO.<br>OF CHANNELS | CHANNELS<br>TO TRIP | MINIMUM<br>CHANNELS<br>OPERABLE | APPLICABLE<br>MODES | ACTION |
|--|--------------------------|---------------------|---------------------------------|---------------------|--------|
| 1. Manual Reactor Trip                               | 2                        | 1                   | 2                               | 1, 2                | 1      |
|  | 2                        | 1                   | 2                               | 3*, 4*, 5*          | 10     |
| 2. Power Range, Neutron Flux                         |                          |                     |                                 |                     |        |
| a. High Setpoint                                     | 4                        | 2                   | 3                               | 1, 2                | 2#     |
| b. Low Setpoint                                      | 4                        | 2                   | 3                               | 1###, 2             | 2#     |
| 3. Power Range, Neutron Flux<br>High Positive Rate   | 4                        | 2                   | 3                               | 1, 2                | 2#     |
| 4. Power Range, Neutron Flux,<br>High Negative Rate  | 4                        | 2                   | 3                               | 1, 2                | 2#     |
| 5. Intermediate Range, Neutron Flux                  | 2                        | 1                   | 2                               | 1###, 2             | 3      |
| 6. Source Range, Neutron Flux                        |                          |                     |                                 |                     |        |
| a. Startup   | 2                        | 1                   | 2                               | 2##                 | 4      |
| b. Shutdown  | 2                        | 1                   | 2                               | 3*, 4*, 5*          | 10     |
| 7. Overtemperature $\Delta T$<br>Four Loop Operation | 4                        | 2                   | 3                               | 1, 2                | 6#     |
| 8. Overpower $\Delta T$<br>Four Loop Operation       | 4                        | 2                   | 3                               | 1, 2                | 6#     |
| 9. Pressurizer Pressure-Low                          | 4                        | 2                   | 3                               | 1                   | 6#**   |

CATAMBA - UNITS 1 &amp; 2

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Amendment No. 39 (Unit 1)  
Amendment No. 31 (Unit 2)

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

- ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.
- ACTION 5 - Delete
- ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:
- a. The inoperable channel is placed in the tripped condition within 6 hours, and
  - b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.
- ACTION 7 - Delete
- ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive status light(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.
- ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.
- ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.
- ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.



TABLE 4.3-1 (Continued)

TABLE NOTATIONS

- \* Only if the Reactor Trip System breakers happen to be closed and the Control Rod Drive System is capable of rod withdrawal.
  - # Above P-9 (Reactor Trip on Turbine Trip Interlock) Setpoint.
  - ## Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.
  - ### Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.
- (1) If not performed in previous 7 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 for 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, 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) With power greater than or equal to the interlock setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive status light.
  - (9) Monthly 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 status light.
  - (10) Setpoint verification is not applicable.
  - (11) At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.
  - (12) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.
  - (13) For Unit 1, CHANNEL CALIBRATION shall ensure that the filter time constant associated with Steam Generator Water Level Low-Low is adjusted to a value less than or equal to 1.5 seconds.

## INSTRUMENTATION

### MOVABLE INCORE DETECTORS

#### LIMITING CONDITION FOR OPERATION

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3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

- a. At least 75% of the detector thimbles,
- b. A minimum of two detector thimbles per core quadrant, and
- c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

- a. Recalibration of the Excore Neutron Flux Detection System, or
- b. Monitoring the QUADRANT POWER TILT RATIO, or
- c. Measurement of  $F_{\Delta H}^N$ , and  $F_Q(Z)$

#### ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

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4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by irradiating each detector used and determining the acceptability of its voltage curve for:

- a. Recalibration of the Excore Neutron Flux Detection system, or
- b. Monitoring the QUADRANT POWER TILT RATIO, or
- c. Measurement of  $F_{\Delta H}^N$ , and  $F_Q(Z)$ .

## INSTRUMENTATION

### SEISMIC INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

---

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

- a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.
- b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

---

4.3.3.3.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above accessible seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01 g shall be restored to OPERABLE status within 24 hours following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. Data retrieved from the triaxial time-history accelerograph shall include a post-event CHANNEL CALIBRATION obtained by actuation of the internal test and calibrate function immediately prior to removing data. CHANNEL CALIBRATION shall be performed immediately after insertion of the new recording media in the triaxial time-history accelerograph recorder. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

## INSTRUMENTATION

### 3/4.3.3.12 BORON DILUTION MITIGATION SYSTEM

#### LIMITING CONDITION FOR OPERATION

---

3.3.3.12 As a minimum, two trains of the Boron Dilution Mitigation System shall be OPERABLE and operating with Shutdown Margin Alarm ratios set at less than or equal to 4 times the steady-state count rate.

APPLICABILITY: MODES 3, 4, AND 5#

#### ACTION:

- (a) With one train of the Boron Dilution Mitigation System inoperable or not operating, restore the inoperable train to OPERABLE status within 48 hours, or
  - (1) suspend all operations involving positive reactivity changes and verify that valve NV-230 is closed and secured within the next hour, or
  - (2) verify two Source Range Neutron Flux Monitors are OPERABLE with Alarm Setpoints less than or equal to one-half decade above the steady-state count rate and verify that the combined flowrate from both Reactor Makeup Water Pumps is less than or equal to 200 gpm (Mode 3) or 80 gpm (Mode 4 or 5) within the next hour.
- (b) With both trains of the Boron Dilution Mitigation System inoperable or not operating, restore the inoperable trains to OPERABLE status within 12 hours, or
  - (1) suspend all operations involving positive reactivity changes and verify that valve NV-230 is closed and secured within the next hour, or
  - (2) verify two Source Range Neutron Flux Monitors are OPERABLE with Alarm Setpoints less than or equal to one-half decade above the steady-state count rate and verify that the combined flow rate from both Reactor Makeup Water Pumps is less than or equal to 200 gpm (Mode 3) or 80 gpm (Mode 4 or 5) within the next hour.
- (c) The provisions of Specification 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

---

4.3.3.12.1 Each train of the Boron Dilution Mitigation System shall be demonstrated OPERABLE by performance of:

- (a) A CHANNEL CHECK at least once per 12 hours,

#Not applicable for Unit 2 until after entering Mode 2 following the first refueling outage.

## INSTRUMENTATION

### SURVEILLANCE REQUIREMENTS (Continued)

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- (b) An ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and
- (c) At least once per 18 months the BDMS shall be demonstrated OPERABLE by:
  - (1) Verifying that each automatic valve actuated by the BDMS moves to its correct position upon receipt of a trip signal, and
  - (2) Verifying each reactor makeup water pump stops, as designed, upon receipt of a trip signal.

4.3.3.12.2 If using the Source Range Neutron Flux Monitors to meet the requirements of Technical Specification 3.3.3.12,

- (a) The monthly surveillance requirements of Table 4.3-1 for the Source Range Neutron Flux Monitors shall include verification that the Alarm Setpoint is less than or equal to one-half decade (square root of 10) above the steady-state count rate.
- (b) The combined flow rate from both Reactor Makeup Water Pumps shall be verified as less than or equal to 200 gpm (Mode 3) or 80 gpm (Mode 4 or 5) at least once per 31 days.

### 3/4.9 REFUELING OPERATIONS

#### 3/4.9.1 BORON CONCENTRATION

##### LIMITING CONDITION FOR OPERATION

---

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met either:

- a. A  $K_{eff}$  of 0.95 or less, or
- b. A boron concentration of greater than or equal to 2000 ppm.

APPLICABILITY: MODE 6.\*

##### ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and 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 its equivalent until  $K_{eff}$  is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive.

##### SURVEILLANCE REQUIREMENTS

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4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

- a. Removing or unbolting the reactor vessel head, and
- b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

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\*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.



## REFUELING OPERATIONS

### 3/4.9.2 INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

3.9.2.1 As a minimum, two trains of the Boron Dilution Mitigation System shall be OPERABLE and operating with Shutdown Margin Alarm Ratios set at less than or equal to 4 times the steady-state count rate, each with continuous indication in the control room.

APPLICABILITY: MODE 6#

#### ACTION:

- (a) With one or both trains of the Boron Dilution Mitigation System inoperable or not operating,
  - (1) immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes, or
  - (2) verify that two Source Range Neutron Flux Monitors are OPERABLE and operating with Alarm Setpoints less than or equal to one-half decade (square root of 10) above the steady-state count rate, each with continuous visual indication in the control room and one with audible indication in the control room and one with audible indication in the containment within the next hour.
- (b) With both trains of the Boron Dilution Mitigation System inoperable or not operating and one of the Source Range Neutron Flux Monitors inoperable or not operating immediately suspend all operations involving core ALTERATIONS or positive reactivity changes.
- (c) With both trains of the Boron Dilution Mitigation System inoperable or not operating and both of the Source Range Neutron Flux Monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.
- (d) The provisions of Specification 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENT

4.9.2.1.1 Each train of the Boron Dilution Mitigation System shall be demonstrated OPERABLE by performance of:

- (a) A CHANNEL CHECK at least once per 12 hours,
- (b) An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS and
- (c) An ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days.

#Not applicable for Unit 2 until after entering Mode 2 following the first refueling outage.

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## 3/4.2 POWER DISTRIBUTION LIMITS

### BASES

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

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

$F_Q(Z)$  Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation  $Z$  divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F_{\Delta H}^N$  Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

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

## POWER DISTRIBUTION LIMITS

### BASES

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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 limit 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  $\pm 3\%$  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  $Q$  value, the Base Load operating procedure restricts the indicated

## POWER DISTRIBUTION LIMITS

### BASES

#### 3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

AFD to relatively small target band and power swings (AFD target band of  $\pm 3\%$ ,  $APL^{ND} \leq \text{power} \leq APL^{BL}$  or 100% Rated Thermal Power, whichever is lower). For Base Load operation, it is expected that the Units will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24 hour waiting period at a power level above  $APL^{ND}$  and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the Base Load procedure. After the waiting period extended Base Load operation is permissible.

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

The limits on heat flux hot channel factor, coolant 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 insure that the limits are maintained provided:

- a. Control rods in a single group move together with no individual rod insertion differing by more than  $\pm 12$  steps, indicated, from the group demand position;
- b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

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## POWER DISTRIBUTION LIMITS

### BASES

#### HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM 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.

$F_{\Delta H}^N$  will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on Figure 3.2-3, Reactor Coolant System flow rate and  $F_{\Delta H}^N$  may be "traded off" against one another (i.e., a low measured Reactor Coolant System flow rate is acceptable if the measured  $F_{\Delta H}^N$  is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of  $F_{\Delta H}^N$  as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for  $F_{\Delta H}^N$  less than or equal to 1.49. This value is used in the various accident analyses where  $F_{\Delta H}^N$  influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed. The rod bow penalty as a function of burnup applied for  $F_{\Delta H}^N$  is calculated with the methods described in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," July 1979, and the maximum rod bow penalty is 2.7% DNBR. Since the safety analysis is performed with plant-specific safety DNBR limits of 1.49 and 1.47 compared to the design DNBR limits of 1.34 and 1.32, respectively, for the typical and thimble cells, there is a 10% thermal margin available to offset the rod bow penalty of 2.7% DNBR.

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  $W(z)$  function for normal operation is provided in the Peaking Factor Limit Report per Specification 6.9.1.9.



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## 3/4.9 REFUELING OPERATIONS

### BASES

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#### 3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for  $K_{eff}$  includes a 1%  $\Delta k/k$  conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron.

#### 3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Boron Dilution Mitigation System ensures that monitoring capability is available to detect changes in the reactivity condition of the core.

#### 3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

#### 3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY of the Reactor Building Containment Purge System ensure that a release of radioactive material within containment will be restricted from leakage to the environment or filtered through the HEPA filters and activated carbon adsorbers prior to release to the atmosphere. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE. Operation of the Reactor Building Containment Purge System and the resulting iodine removal capacity are consistent with the assumption of the safety analysis. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

## ADMINISTRATIVE CONTROLS

### SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

### MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555, with a copy to the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

### PEAKING FACTOR LIMIT REPORT

6.9.1.9 The  $W(z)$  functions for RAOC and Base Load operation and the value for  $APL^{ND}$  (as required) shall be provided to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555 with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation.

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.