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

February 16, 1988

Docket Nos.: 50-413
and 50-414

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Issuance of Amendment No. 39 to Facility Operating License NPF-35 and Amendment No. 31 to Facility Operating License NPF-52 - Catawba Nuclear Station, Units 1 and 2 (TACS 66636/66637)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 39 to Facility Operating License NPF-35 and Amendment No. 31 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated November 13, 1987, and supplemented December 11, 1987, January 15 and 20, 1988.

The amendments modify the Technical Specifications to ensure that plant operation is consistent with the design and safety evaluation conclusions of the Unit 2 cycle 2 reload safety evaluation and to reflect the addition of the Boron Dilution Mitigation System for Unit 2.

A copy of the related safety evaluation supporting Amendment No. 39 to Facility Operating License NPF-35 and Amendment No. 31 to Facility Operating License NPF-52 is enclosed.

Notice of issuance of amendments will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

Kahtan N. Jabbour

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

Enclosures:
1. Amendment No. 39 to NPF-35
2. Amendment No. 31 to NPF-52
3. Safety Evaluation

cc w/enclosures: See next page

DATED February 25, 1998

AMENDMENT NO. 152 TO FACILITY OPERATING LICENSE NPF-35, CATAWBA UNITS 1 AND 2
AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NPF-52, CATAWBA UNITS 1 AND 2

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February 16, 1988

Docket Nos.: 50-413
and 50-414

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Issuance of Amendment No. 39 to Facility Operating License NPF-35 and Amendment No. 31 to Facility Operating License NPF-52 - Catawba Nuclear Station, Units 1 and 2 (TACS 66755/66756)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 39 to Facility Operating License NPF-35 and Amendment No. 31 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated November 13, 1987, and supplemented December 11, 1987, January 15 and 20, 1988.

The amendments modify the Technical Specifications to ensure that plant operation is consistent with the design and safety evaluation conclusions of the Unit 2 cycle 2 reload safety evaluation and to reflect the addition of the Boron Dilution Mitigation System for Unit 2.

A copy of the related safety evaluation supporting Amendment No. 39 to Facility Operating License NPF-35 and Amendment No. 31 to Facility Operating License NPF-52 is enclosed.

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Sincerely,

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

Enclosures:
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2. Amendment No. 31 to NPF-52
3. Safety Evaluation

cc w/enclosures: See next page

	KNJ	KNJ
PDII-3/DRPI/II	PDII-3/DRPI/II	PDII-3/DRPI/II
MRood/sw	K Jabbour	Acting PD
02/4 /88	02/10 /88	02/10 /88

Mr. H. B. Tucker
Duke Power Company

Catawba Nuclear Station

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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:

(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 KNJ

PDII-3/DRPI/II	PDII-3/DRPI/II	OGC-WF
MRood/mac	KJabbour	*MYoung
02/ 4/88	02/10/88	01/29/88

KNJ
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

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	KNJ PDII-3/DRPI/II	OGC-WF
MRood/mac	KJabbour	MYoung*
02/ 4/88	01/15/88	01/ /88
KNJ PDII-3/DRPI/II		
Acting PD		
02/10/88		

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.

Amended Page	Overleaf Page
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 ΔT 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 ΔT 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%.

SURVEILLANCE REQUIREMENTS (Continued)

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 ± 1% Δk/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.

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% Δk/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

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% Δk/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.

CONTROL BANK INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

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

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

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.

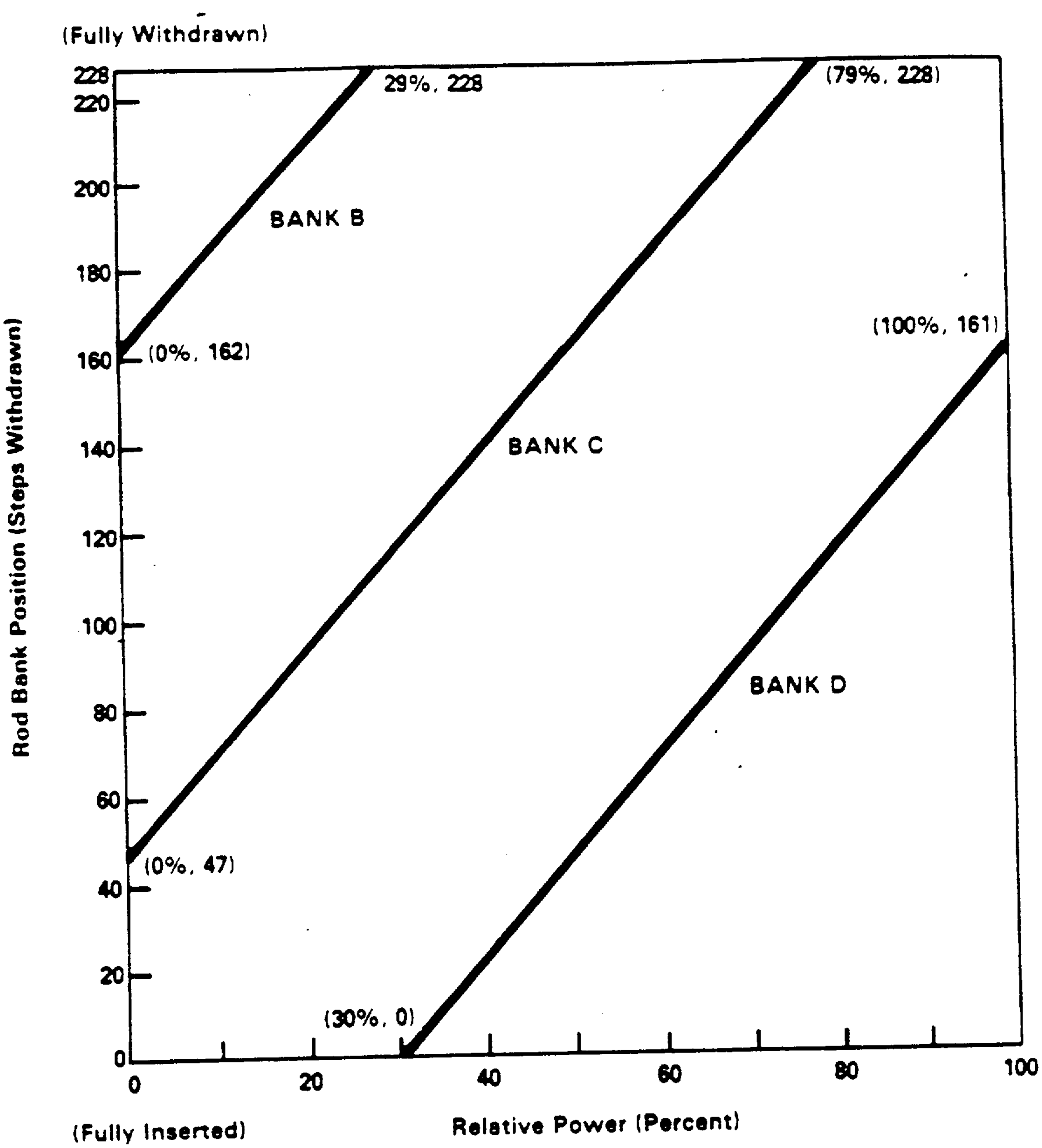

FIGURE 3.1-1

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

This page intentionally deleted.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

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

a. the allowed operational space defined by Figure 3.2-1 for RAOC operation, or

b. within a ±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^{ND}** 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^{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 Exceptions Specification 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 6.9.1.9.

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.

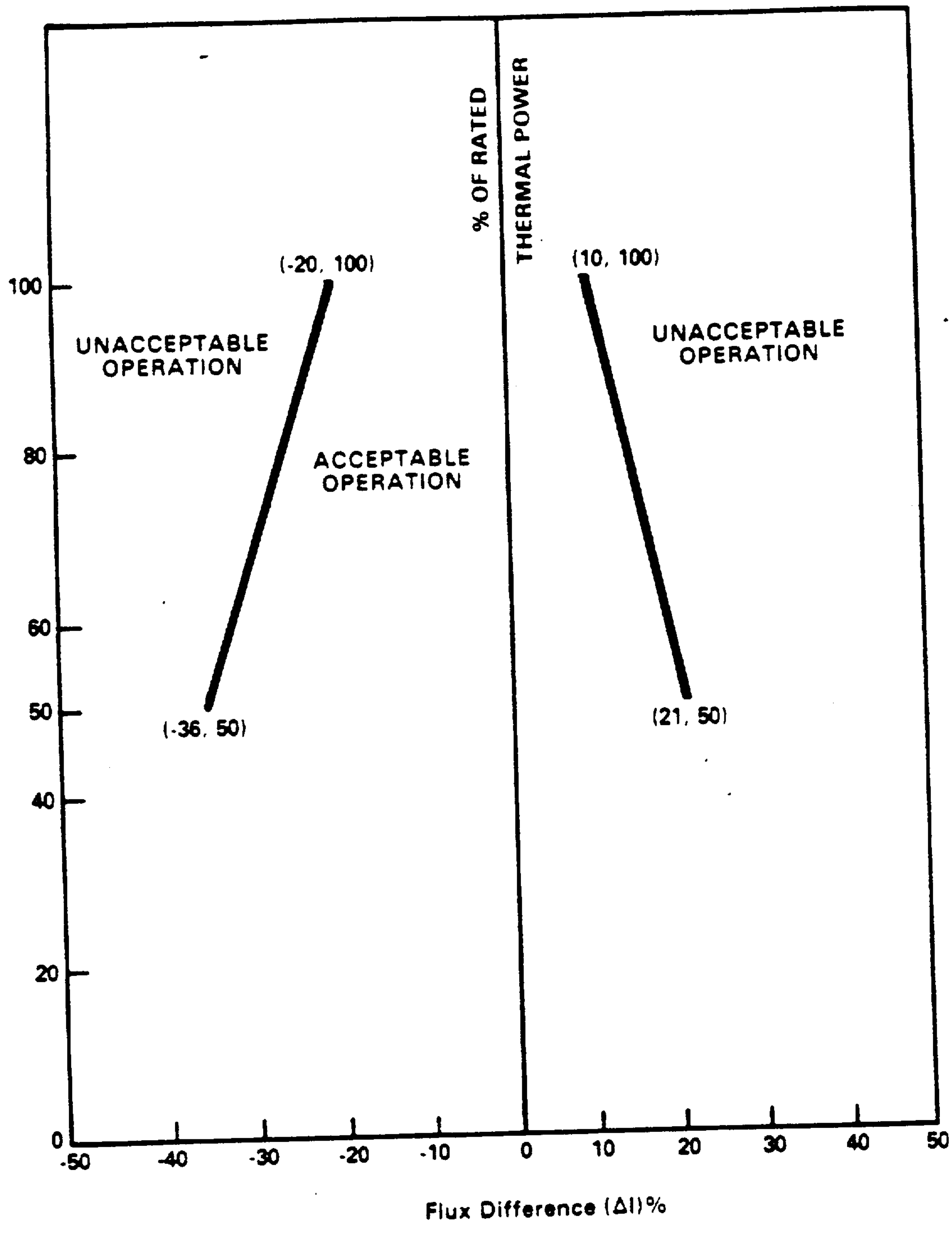

FIGURE 3.2-1
AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

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

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:

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

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

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 (value of K_4) have been reduced at least 1% (in ΔT 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.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

c. Satisfying the following relationship:

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

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

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 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.

*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

$$\underset{\text{over } z}{\text{maximum}} \quad \frac{F_Q^M(z)}{K(z)}$$

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

$$\underset{\text{over } z}{\text{maximum}} \quad \frac{F_Q^M(z)}{K(z)} \quad \text{is not increasing.}$$

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:

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

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

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 deter mined 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 permited 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 ±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:

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

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 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.

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

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

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 4.2.2.4c, or

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

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

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

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:

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

g. The limits specified in 4.2.2.4c., 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. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE 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 High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, 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 ΔT Four Loop Operation	4	2	3	1, 2	6#
8. Overpower ΔT Four Loop Operation	4	2	3	1, 2	6#
9. Pressurizer Pressure-Low	4	2	3	1	6#**

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.

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

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 $F^N_{\Delta H}$, 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

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 $F^N_{\Delta H}$, and $F_Q(Z)$.

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.

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 followng the first refueling outage.

SURVEILLANCE REQUIREMENTS (Continued)

(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

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.

*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.

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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BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the calculated DNBR in the core 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;

$F^N_{\Delta H}$ 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; 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.

BASES

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 ±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

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 ±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 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 ΔI power operating space (for RAOC operation), or 2) outside the allowed ΔI 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 ± 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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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.

$F^N_{\Delta H}$ 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 $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured Reactor Coolant System flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure 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.

R as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ 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. The rod bow penalty as a function of burnup applied for $F^N_{\Delta H}$ 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 $F^M_Q(z)$ 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 6.9.1.9.

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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.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NPF-35

AND AMEDMENT NO. 31 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY ET AL.

DOCKET NOS. 50-413 AND 50-414

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

INTRODUCTION

By letter dated November 13, 1987 (Ref. 1), Duke Power Company, et al., (the licensee) requested Changes to the Technical Specifications (TSs) for Catawba Nuclear Station Units 1 and 2, to reflect the Unit 2 refueling and the addition of the Boron Dilution Mitigation System for Unit 2. In addition changes to TSs 4.3.3.12.1(b); 3.9.2.1, Actions (a)(2) and (d); and 4.9.1.3 are requested for both units. A supplemental letter dated December 11, 1987 (Ref. 2) provided a discussion of the Justification and No Significant Hazards Considerations. Additional information and justification were provided in letters dated January 15 and 20, 1988 (Refs. 9 and 10).

The substance of the changes noticed in the Federal Register on December 30, 1987 and the proposed no significant hazards determination were not affected by the licensee's letters dated January 15 and 20, 1988 which clarified certain aspects of the request.

EVALUATION

A. Unit 2 Cycle 2 Reload

1. General Design

The Catawba Unit 2, Cycle 2, reactor core contains 193 Optimized Fuel Assemblies. During the Cycle 1/2 refueling 64 Region 1 fuel assemblies will be replaced with 64 Region 4 fuel assemblies. The Region 4 fuel is very similar to that used in Regions 1, 2, and 3. Region 4 fuel assemblies have a smaller rod plenum spring than those used in Regions 1, 2, and 3. This new spring design is being generally incorporated by Westinghouse and the justification was submitted in Reference 3. The Region 4 fuel has been designed according to the fuel performance model in WCAP 8785 (Ref. 4). The fuel is designed and operated so that clad flattening will not occur as provided by the Westinghouse model in WCAP 8377 (Ref. 5). For all fuel regions, the fuel rod internal pressure design basis, which is discussed and shown acceptable in WCAP-8964 (Ref. 6) is satisfied.

The licensee provided a Reload Safety Evaluation (RSE) for Catawba 2, Cycle 2, as an attachment to Reference 1. The RSE presents a Cycle-specific evaluation for Cycle 2 which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was performed utilizing the approved reload design methods of WCAP-9273-U-A (Ref. 7).

2. Nuclear Design

The Cycle 2 Core loading is designed to meet an $[F_Q(Z)xP]$ ECCS limit of less than or equal to 2.32xK(Z). Adherence to the F_Q limit is obtained by using the F_Q TS surveillance described in WCAP-10217-A (Ref. 8). F_Q surveillance is part of the Relaxed Axial Offset Control (RAOC) and replaces the previous F_{xy} surveillance by comparing a measured F_Q limit. This provides a more convenient form of assuring plant operation below the F_Q limit while retaining the intent of using a measured parameter to verify operation below TS limits. The above discussion is consistent with Reference 8 which was approved. Thus, the staff finds that the TS change to F_Q surveillance is acceptable.

RAOC will be employed in Cycle 2 to enhance operational flexibility during non-steady state operation. RAOC makes use of available margin by expanding the allowable delta I band, particularly at reduced power. ROAC is described in Reference 8 and was approved by the staff. Thus, it is acceptable for use in Catawba Unit 2.

During operation at or near steady state equilibrium conditions, core peaking factors are significantly reduced due to the limited amount of xenon skewing possible under these operating conditions. The licensee proposes to use Base Load TSs to recognize this reduction in core peaking factors. The proposed Base Load TSs are identical to those that the staff has previously approved for McGuire Units 1 and 2, and Catawba Unit 1 and are therefore acceptable.

The RSE provides a table of Cycle 2 kinetics characteristics which are compared with the current limits based on previously approved accident analyses. The RSE also provides a table showing the results of the calculated Cycle 2 control rod worths and requirements at the most limiting condition during the cycle (end-of-life). These results include a standard 10% allowance for calculational uncertainty. From this information, the staff concludes that sufficient control rod worth will be available to provide the required shutdown margin for Cycle 2 operation. Control rod insertion limits were increased for less than 100% power for Cycle 2. Since the required shutdown margin is maintained, the TS change proposed to reflect the increased insertion is acceptable.

3. Thermal and Hydraulic Design

The thermal hydraulic methodology, DNBR correlation and core DNB limits used for Cycle 2 are consistent with the current licensing basis described in the FSAR and approved by the staff.

The power distributions produced by the cycle-specific RAOC analysis were analyzed for normal operation and Condition II events. Limits on the allowable operating flux difference as a function of power level from these considerations were found to be less restrictive than those resulting from LOCA F_Q considerations. The Condition II analyses generate DNB core limits and resultant overtemperature delta T setpoints. These generated a change to the $F(\Delta I)$ function in the TSs. The change is acceptable because it results from cycle-specific calculations using approved methods (Refs. 7 and 8). Therefore, the staff concludes that the Cycle 2 thermal-hydraulic analysis is acceptable.

4. Accident Analysis

The effects of the reload on the design basis and postulated accidents analyzed in the FSAR were examined. In all cases it was found that the effects were accommodated within the conservatism of the initial assumption used in the previous applicable safety analysis as well as those performed in support of the RTD Bypass removal and the UHI deletion (Refs. 1, 2 and 9). A core reload can affect accident analysis input parameters through control rod worths, core peaking factors and core kinetic characteristics. The Cycle 2 parameters in each of these areas were examined and found to be within the bounds of the current limits. Therefore, the staff concludes that the accident analysis is acceptable.

5. Technical Specification Changes

The Technical Specification changes for the Unit 2 Cycle 2 Reload are:

1. RAOC and Axial Flux Difference Limits
2. F_Q Surveillance
3. Base Load TSs
4. Rod Insertion Limits
5. $OT\Delta T$ f_1 (ΔI)

Acceptability of items 1 - 4 was discussed in Section 2, Nuclear Design. Acceptability of item 5 was discussed in Section 3, thermal and hydraulic design. The proposed changes are for Unit 2 only but Unit 1 is included only administratively because the TSs for both units are combined in one document. The revisions to the bases are also acceptable.

B. Boron Dilution Mitigation System

1. Introduction

The Boron Dilution Mitigation System (BDMS) which is being installed in Unit 2 is the same as the BDMS which was installed in Unit 1. The BDMS was described in letters dated June 6, 1986 and September 9, 1986 (Refs. 11 and 12).

2. Technical Specification Changes

The changes for Unit 2 which deal with the (BDMS) are to TSs 4.1.1.1.3; 4.1.1.1.4; 4.1.1.2.2; Table 3.3-1, item 6.b; Table 3. 3-1, Action 5; Table 4.3-1, Note (9); 3/4.3.3.12; and 3/4.9.2. Changes to TSs 4.3.3.12.1(b); 3.9.2.1, Actions (a)(2) and (d); and 4.9.1.3 apply to both Units. Each change is discussed below.

The changes that apply to Unit 2 only are identical to those approved for Unit 1 TSs when the (BDMS) was installed in that Unit. The licensee requested that these changes not apply to Unit 2 until after the BDMS system has been calibrated, tested and declared operable. Furthermore, licensee stated (Ref. 10) that all the TSs applicable to boron dilution accidents which are to be deleted, will be adminstratively maintained in this interim period. The staff finds this acceptable.

TS 4.3.3.12.1(b)

This TS will be deleted because it is required only prior to Mode 2 but the specification itself is not applicable in Modes 1 and 2. The staff finds this change acceptable.

TS 3.9.2.1, Actions (a)(2) and (d)

This change to Action (a)(2)is an editorial change which deletes a phrase "and control room" which appeared twice in the sentence. Thus it is acceptable. The addition of Action (d) would allow the plant to change modes if the BDMS is inoperable. This statement already appears in TS 3.3.3.12 which covers all other applicable modes.

TS 4.9.1.3

This TS verifies that potential boron dilution flow paths are isolated when the unit is in Mode 6. The deletion of TS 4.9.1.3 is acceptable because the BDMS provides for automatic isolation of potential boron dilution flow paths.

ENVIRONMENTAL CONSIDERATION

These amendments involve changes to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The staff has determined that the amendments involve no significant increase in the ammounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational exposures. The NRC staff has made a determination that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (52 FR 49225) on December 30, 1987. The Commission consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

We have 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

(1) Letter form Hal B. Tucker (Duke Power Company) to NRC, November 13, 1987

(2) Letter from Hal B. Tucker (Duke Power Company) to NRC, December 11, 1987

(3) Letter from E. P. Rahe Jr. (Westinghouse) to L. E. Phillps (NRC) April 12, 1984, NS-EPR-2893, Subject: Fuel Handling Load Curtain (6g vs 4g)

(4) Muller, J. V. (Ed.) "Improved Analytical Model used in Westinghouse Fuel Rod Design Computations, "WCAP-8785, October 1976

(5) George, R. A., (et al.), "Revised Clad Flattening Model," WCAP-8377, July 1977

(6) Risher, D. H., (et al.), "Safety Analyses for the Reused Fuel Rod Internal Pressure Design Basis," WCAP-8964, June 1977

(7) Bordelon, F. M., (et al.), "Westinghouse Reload Safety Evaluation Methodology", WCAP-9273 U-A, July 1985

(8) Muller, R. W. , (et al.), "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification, "WCAP-10217-A, June 1983

(9) Letter from Hal B. Tucker (Duke Power Company) to NRC, January 15, 1988

(10) Letter from Hal B. Tucker (Duke Power Company) to NRC, January 20, 1988

(11) Letter from Hal B. Tucker (Duke Power Company) to Harold Denton (NRC), June 6, 1986

(12) Letter from Hal B. Tucker (Duke Power Company) to Harold Denton (NRC), September 9, 1986

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