October 1, 1986

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. 14 to Facility Operating License NPF-35 and Amendment No. 6 to Facility Operating License NPF-52 - Catawba Nuclear Station, Units 1 and 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 14 to Facility Operating License NPF-35 and Amendment No. 6 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 July 15, 1986, and supplemented July 24, 1986.

These amendments modify Technical Specifications related to application of a positive moderator temperature coefficient and to reflect the Cycle 2 refueling for Unit 1.

A copy of the related safety evaluation supporting Amendment No. 14 to facility Operating License NPF-35 and Amendment No. 6 to Facility Operating License NPF-52 is enclosed.

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

Sincerely,

/s/

Kahtan Jabbour, Project Manager
PWR Project Directorate #4
Division of PWR Licensing-A

Enclosures:
1. Amendment No. 14 to NPF-35
2. Amendment No. 6 to NPF-52
3. Safety Evaluation

cc w/encl:
See next page

PWR#4/DPWR-A	PWR#4/DPWR-A	PWR#4/DPWR-A
MDuncan/rad	KJabbour	BJYoungblood
09/17/86	09/17/86	09/30/86

8610150351 861001
PDR ADOCK 05000413
P PDR

Mr. H. B. Tucker
Duke Power Company

Catawba Nuclear Station

cc:
William L. Porter, Esq.
Duke Power Company
P.O. Box 33189
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esq.
Bishop, Liberman, Cook, Purcell
and Reynolds
1200 Seventeenth Street, N.W.
Washington, D. C. 20036

North Carolina MPA-1
Suite 600
3100 Smoketree Ct.
P.O. Box 29513
Raleigh, North Carolina 27626-0513

Mr. C. D. Markham
Power Systems Division
Westinghouse Electric Corp.
P.O. Box 355
Pittsburgh, Pennsylvania 15230

NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 33515

County Manager of York County
York County Courthouse
York South Carolina 29745

Richard P. Wilson, Esq.
Assistant Attorney General
S.C. Attorney General's Office
P.O. Box 11549
Columbia, South Carolina 29211

Piedmont Municipal Power Agency
100 Memorial Drive
Greer, South Carolina 29651

Mark S. Calvert, Esq.
Bishop, Liberman, Cook,
Purcell & Reynolds
1200 17th Street, N.W.
Washington, D. C. 20036

Brian P. Cassidy, Regional Counsel
Federal Emergency Management Agency,
Region I
J. W. McCormach POCH
Boston, Massachusetts 02109

North Carolina Electric Membership
Corp.
3333 North Boulevard
P.O. Box 27306
Raleigh, North Carolina 27611

Saluda River Electric Cooperative,
Inc.
P.O. Box 929
Laurens, South Carolina 29360

Senior Resident Inspector
Route 2, Box 179N
York, South Carolina 29745

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission,
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

Mr. Heyward G. Shealy, Chief
Bureau of Radiological Health
South Carolina Department of Health
and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Karen E. Long
Assistant Attorney General
N.C. Department of Justice
P.O. Box 629
Raleigh, North Carolina 27602

Spence Perry, Esquire
Associate General Counsel
Federal Emergency Management Agency
Room 840
500 C Street
Washington, D. C. 20472

Mr. Michael Hirsch
Federal Emergency Management Agency
Office of the General Counsel
Room 840
500 C Street, S.W.
Washington, D. C. 20472

October 1, 1986

AMENDMENT NO. 14 TO FACILITY OPERATING LICENSE NPF-35 -
CATAWBA NUCLEAR POWER STATION, UNIT 1
AMENDMENT NO. 6 TO FACILITY OPERATING LICENSE NPF-52 -
CATAWBA NUCLEAR POWER STATION, UNIT 2

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B. J. Youngblood
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B. Grimes
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ACRS (10)
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NThompson
EButcher
M. Dunenfeld

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. 14
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 July 15, 1986, and supplemented July 24, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's 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 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 license amendment will not be inimical to the common defense and security or to the health and safety of the public;

   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 and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 14, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this 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

/S/

Kahtan Jabbour, Project Manager
PWR Project Directorate No. 4
Division of PWR Licensing-A

Attachment:
Technical Specification Changes

Date of Issuance: October 1, 1986

PWR#4/DPWR-A	PWR#4/DPWR-A	OGC-Beth	PWR#4/DPWR-A
MDuncan/rad	KJabbour		BJYoungblood
09/17/86	09/17/86	09/24/86	09/30/86

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

DUKE POWER COMPANY

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
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 Electric Membership Corporation and Piedmont Municipal Power Agency, (the licensee) dated July 15, 1986, and supplemented July 24, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's 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 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 license amendment will not be inimical to the common defense and security or to the health and safety of the public;

   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 and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 6 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this 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

/S/

Kahtan Jabbour, Project Manager
PWR Project Directorate No. 4
Division of PWR Licensing-A

Attachment:
Technical Specification Changes

Date of Issuance: October 1, 1986

PWR#4/DPWR-A	PWR#4/DPWR-A	OGC-Beth	PWR#4/DPWR-A
MDuncan/rad	KJabbour	Johnson	BJYoungblood
09/17/86	09/17/86	09/24/86	09/30/86

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND TO

LICENSE AMENDMENT NO.

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 is identified by Amendment number and contains vertical lines indicating the areas of change. The corresponding overleaf pages is also provided to maintain document completeness.

Amended Page	Overleaf Page
2-8	2-7
3/4 1-4	3/4 1-3
3/4 1-5	
3/4 1-5a	3/4 1-6
3/4 1-21	
3/4 1-22	
3/4 1-23	
3/4 2-1	
3/4 2-2	
3/4 2-3	
3/4 2-4	
3/4 2-4a	
3/4 2-4b	
3/4 2-4c	
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	
3/4 2-7e	
3/4 2-7f	3/4 2-8
B 3/4 1-2	B 3/4 1-1
B 3/4 2-1	
B 3/4 2-2	
B 3/4 2-2a	
B 3/4 2-3	
B 3/4 2-4	
B 3/4 2-4a	
6-19	6-20

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT = Measured ΔT by RTD Manifold Instrumentation;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constant utilized in the lag compensator for ΔT, τ_3 = 0;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_1 = 1.411;

K_2 = $0.02401/^{\circ}F$;

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 28 s, τ_5 = 4 s;

T = Average temperature, $^{\circ}F$;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0;

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% (Unit 1) and between -43% and -6.5% (Unit 2), $f_1(\Delta I) = 0$, 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% (Unit 1) and -43% (Unit 2), the ΔT Trip Setpoint shall be automatically reduced by 3.151% (Unit 1) and 2% (Unit 2) 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%.

REACTIVITY CONTROL SYSTEMS

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% $\Delta k/k$.

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

4.1.1.2.1 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.

4.1.1.2.2 At least once per 18 months, each Reactor Makeup Water pump shall be demonstrated OPERABLE by verifying a flow rate of less than or equal to 120 gpm. At least once per 31 days, one Reactor Makeup Water pump shall be demonstrated inoperable by verifying that the motor circuit breaker is secured in the open position.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than the limits shown in Figure 3.1-0;

b. Less negative than -4.1×10^{-4} $\Delta k/k/°F$ for all the rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only#.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only#.

ACTION:

a. With the MTC more positive than the limits shown in Figure 3.1-0, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the limits shown in Figure 3.1-0 within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

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

#See Special Test Exceptions Specification 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the limit of Figure 3.1-0 prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to -3.2×10^{-4} $\Delta k/k/^{\circ}F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.2×10^{-4} $\Delta k/k/^{\circ}F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

REACTIVITY CONTROL SYSTEMS

CONTROL BANK INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figures 3.1-1a (Unit 1) and 3.1-1b (Unit 2).

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using 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.

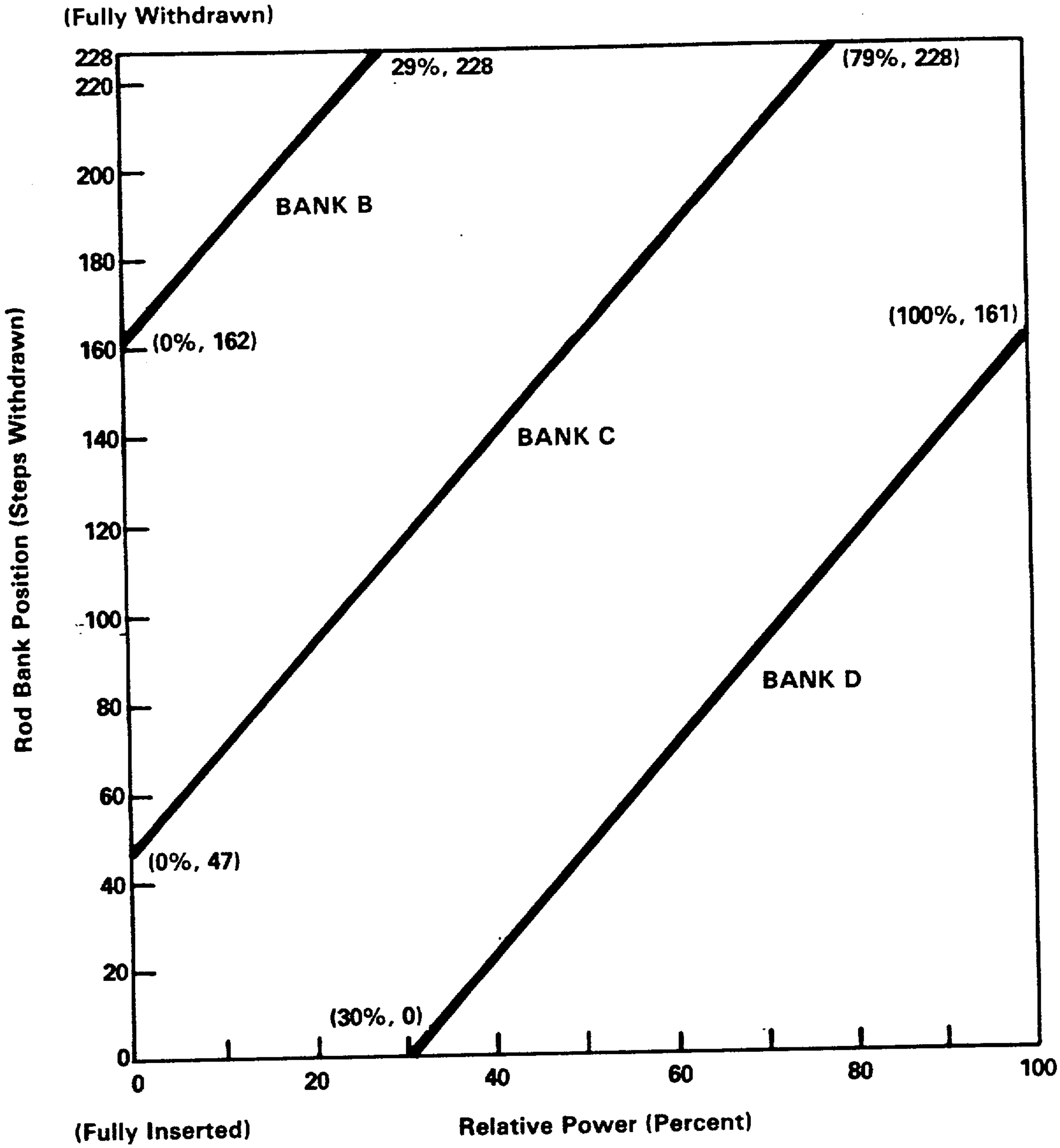

FIGURE 3.1-1a

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION (UNIT 1)

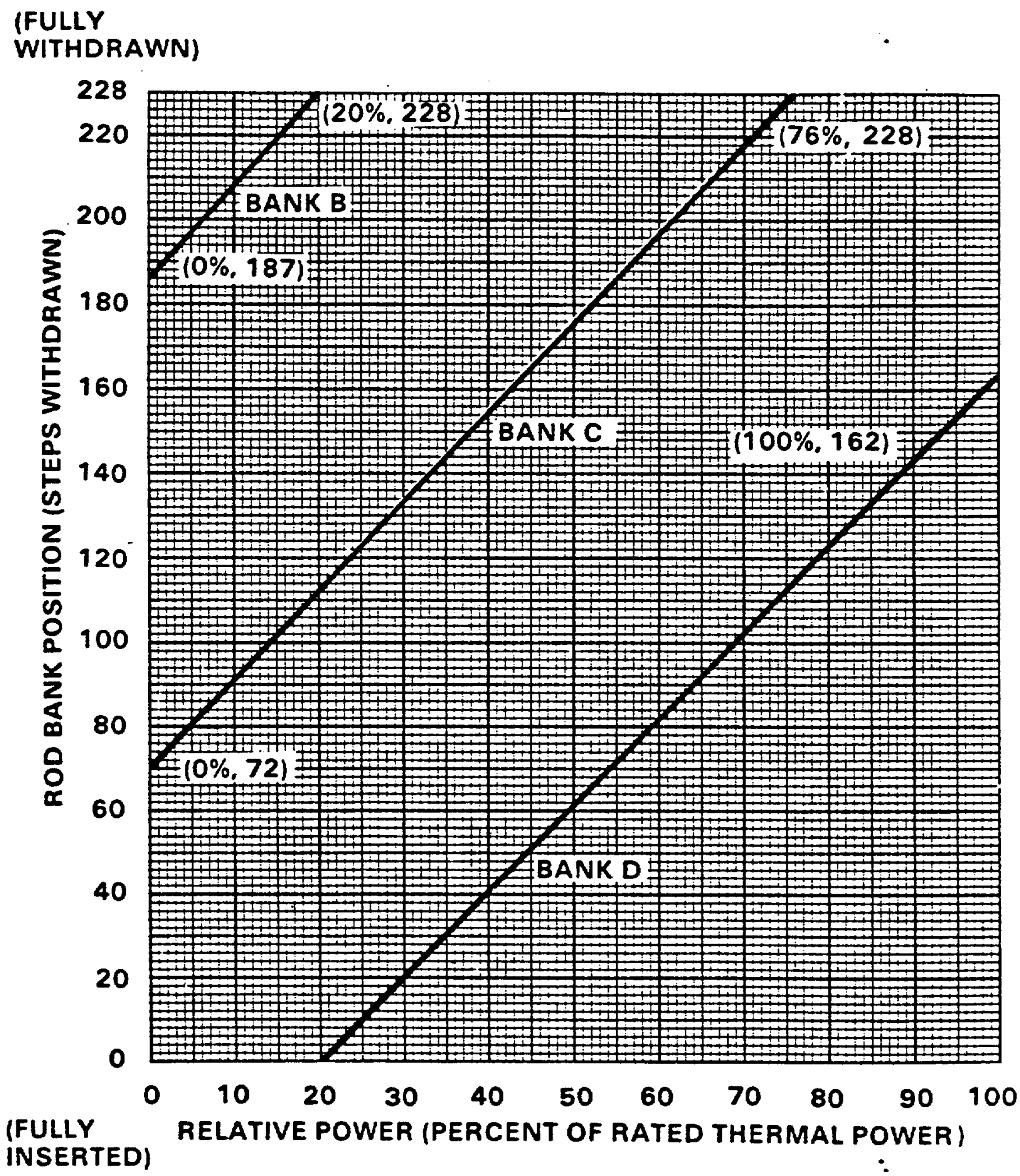

FIGURE 3.1-1b

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION (Unit 2)

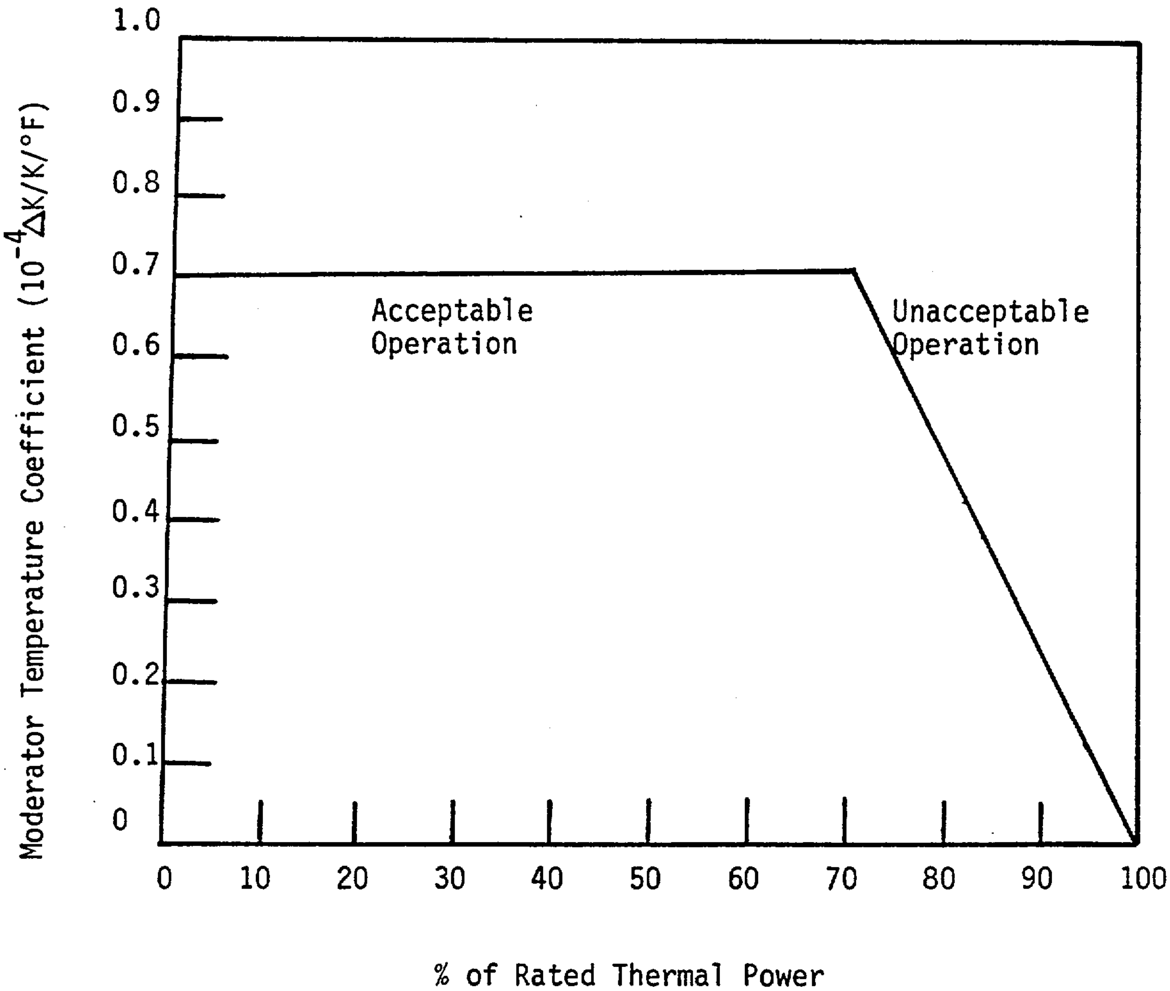

FIGURE 3.1-0

MODERATOR TEMPERATURE COEFFICIENT VS. POWER LEVEL

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 551°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System Temperature (T_{avg}) shall be determined to be greater than or equal to 551°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

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

#See Special Test Exceptions Specification 3.10.3.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD) (UNIT 1)

LIMITING CONDITION FOR OPERATION

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

a. the allowed operational space defined by Figure 3.2-1a 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 (Unit 1).*

ACTION:

a. For RAOC operation with the indicated AFD outside of the Figure 3.2-1a limits,

   1. Either restore the indicated AFD to within the Figure 3.2-1a 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 3.2-1a 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

4.2.1.1.1 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.

4.2.1.1.2 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.

4.2.1.1.3 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.

4.2.1.1.4 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.

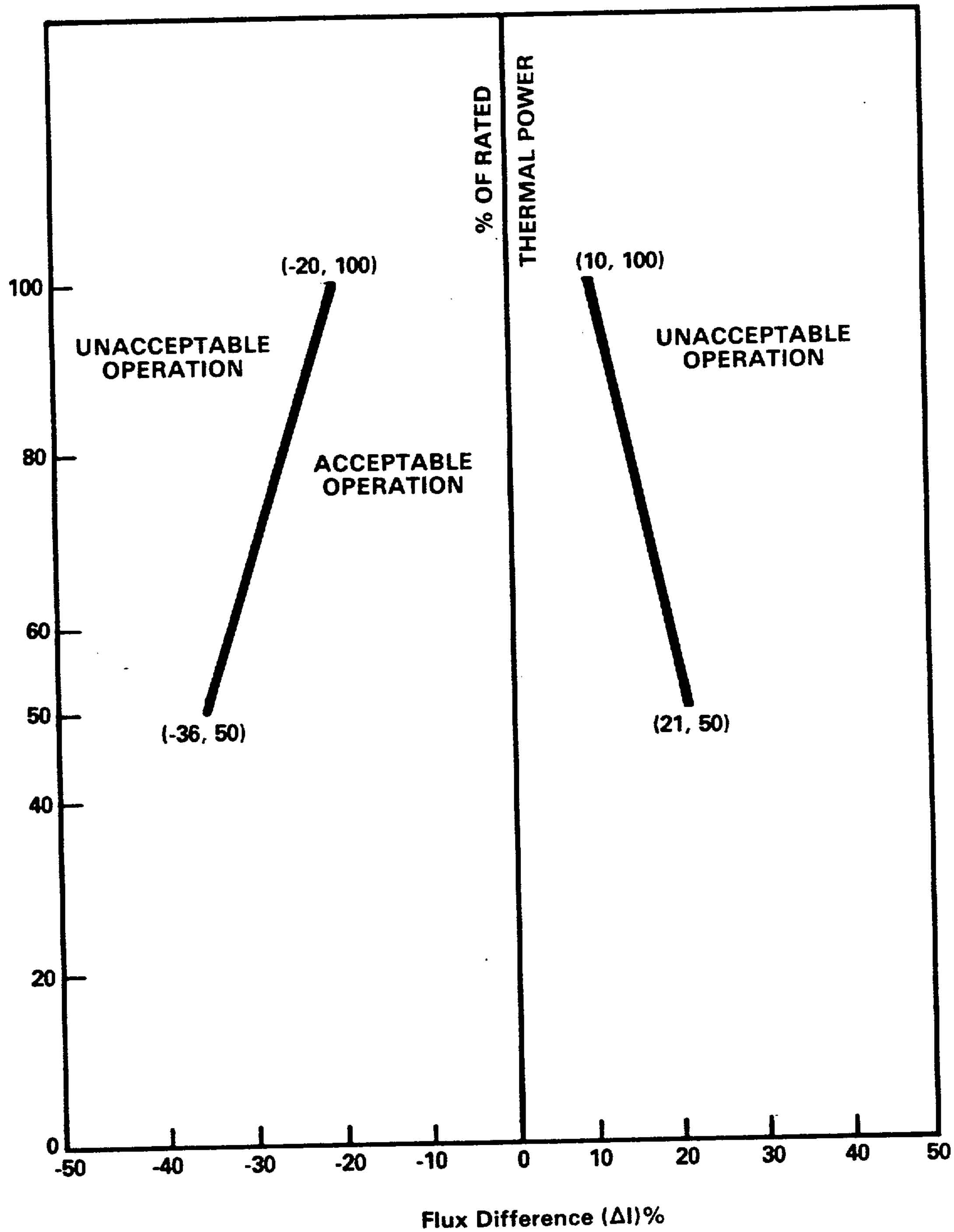

FIGURE 3.2-1a

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER (UNIT 1)

AXIAL FLUX DIFFERENCE (UNIT 2)

LIMITING CONDITION FOR OPERATION

3.2.1.2 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the following target band (flux difference units) about the target flux difference:

a. ±5% for Cycle 1 core average accumulated burnup of less than or equal to 5000 MWD/MTU;

b. +3%, -9% for Cycle 1 core average accumulated burnup of greater than 5000 MWD/MTU; and

c. +3%, -12% for subsequent cycles.

The indicated AFD may deviate outside the above required target level at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1b and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER (Unit 2).*

ACTION:

a. With the indicated AFD outside of the above required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes, either:

   1. Restore the indicated AFD to within the target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation times during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1b and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

   1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

*See Special Test Exceptions Specification 3.10.2.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. The Power Range Neutron Flux* - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

c. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the above required target band.

SURVEILLANCE REQUIREMENTS

4.2.1.1.2 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% 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.

4.2.1.2.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

---

*Surveillance testing of the Power Range Neutron Flux Channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1b. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

SURVEILLANCE REQUIREMENTS (Continued)

b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.2.3 The target 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.

4.2.1.2.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to Specification 4.2.1.2.3 above or by linear interpolation between the most recently measured value and 0% at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

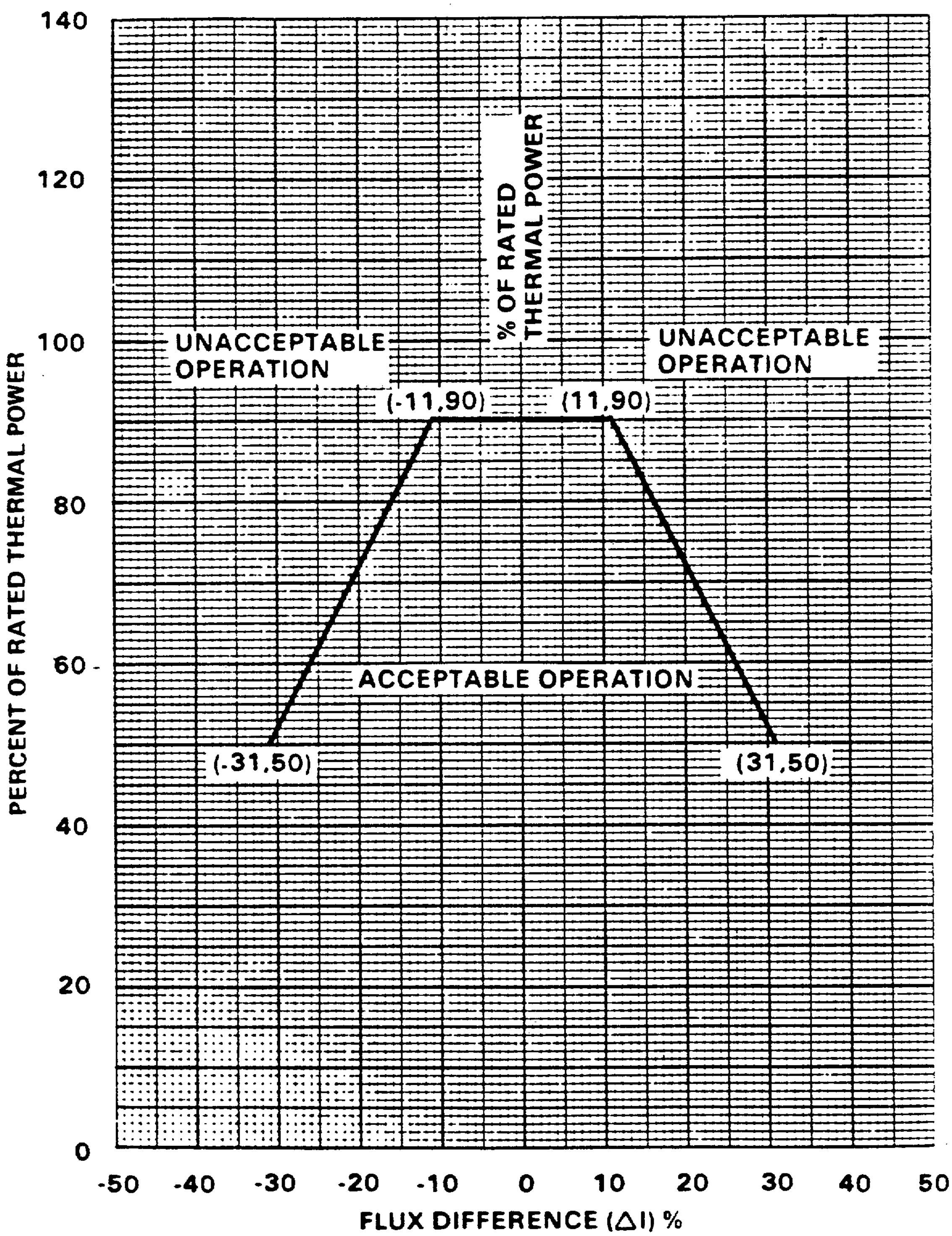

FIGURE 3.2-1b

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER (Unit 2)

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$ (Unit 1)

LIMITING CONDITION FOR OPERATION

3.2.2.1 $F_Q(Z)$ 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 (Unit 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

4.2.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.1.2 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 3.2.2.1 are satisfied.

c. Satisfying the following relationship:

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

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

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

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

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1) $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.1.2c., 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 4.2.2.1.2.c. 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 3.2-1a by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.1.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

b) Comply with the requirements of Specification 3.2.2.1 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

c) Verify that the requirements of Specification 4.2.2.1.3 for Base Load operation are satisfied and enter Base Load operation.

SURVEILLANCE REQUIREMENTS (Continued)

g. The limits specified in Specifications 4.2.2.1.2c., 4.2.2.1.2e., and 4.2.2.1.2f., 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.

4.2.2.1.3 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 4.2.2.1.2 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 4.2.2.1.4. 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 4.2.2.1.3.a shall be satisfied before re-entering Base Load operation.

4.2.2.1.4 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 3.2.2.1 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 4.2.2.1.3 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 4.2.2.1.4.c, 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 4.2.2.1.4c above not being satisfied, either of the following actions shall be taken:

1. Place the core in an equilibrium condition where the limit in 4.2.2.1.2c is satisfied, and remeasure $F_Q^M(Z)$, or

SURVEILLANCE REQUIREMENTS (Continued)

2. Comply with the requirements of Specification 3.2.2.1 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 4.2.2.1.4c., 4.2.2.1.4e., and 4.2.2.1.4f. 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.

4.2.2.1.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.1.2 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.

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$ (Unit 2)

LIMITING CONDITION FOR OPERATION

3.2.2.2 $F_Q(Z)$ 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 (Unit 2).

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

4.2.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2.2 F_{xy} 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_{xy} 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,

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in Specification 4.2.2.2.2b., above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2.2e. and f., below, and

2) The relationship:

$$F_{xy}^L = F_{xy}^{RTP} [1+0.2(1-P)],$$

Where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1) When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L either:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

SURVEILLANCE REQUIREMENTS (Continued)

2) When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.9;

f. The F_{xy} limits of Specification 4.2.2.2.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

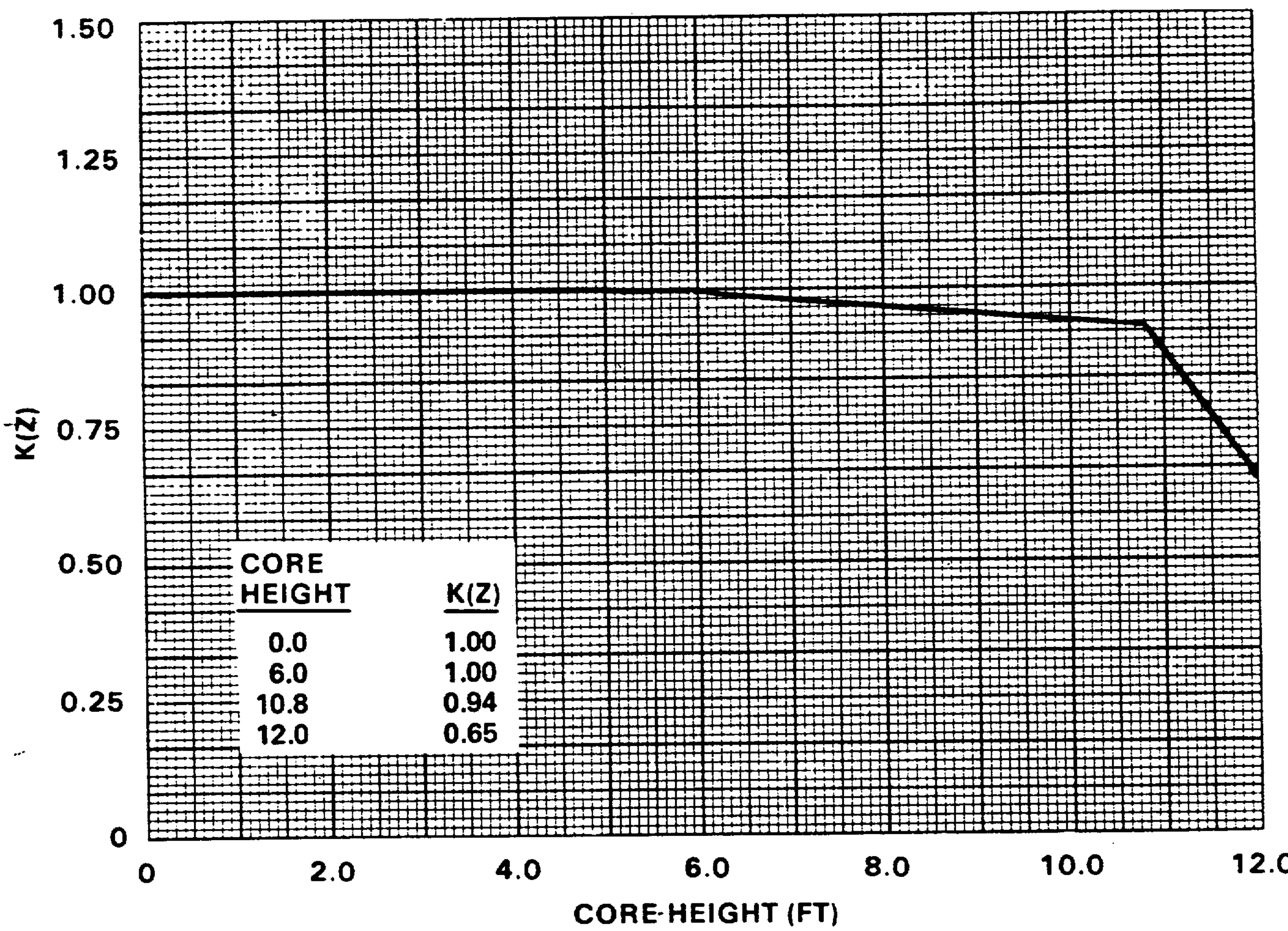

FIGURE 3.2-2

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

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

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

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, boron concentration, and T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled Reactor Coolant System cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

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

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections

MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.1×10^{-4} $\Delta k/k/°F$. The MTC value of -3.2×10^{-4} $\Delta k/k/°F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -4.1×10^{-4} $\Delta k/k/°F$.

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

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the P-12 interlock is above its setpoint, (4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (5) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

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

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

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

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z (Unit 2 only).

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.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that Unit 2 will be operated with the AFD within the target band required by Specification 3.2.1.2 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1b while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

For Unit 2, provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least two of four or two of three OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band for Unit 2.

For Unit 1 at power levels below APL^{ND}, the limits on AFD are defined by Figure 3.2-1a, 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.

For Unit 1 at power levels greater than APL^{ND}, two modes of operation are permissible; 1) RAOC, the AFD limit of which are defined by Figure 3.2-1a, 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

AXIAL FLUX DIFFERENCE (Continued)

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 Unit 1 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.

For Unit 1 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.

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

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;

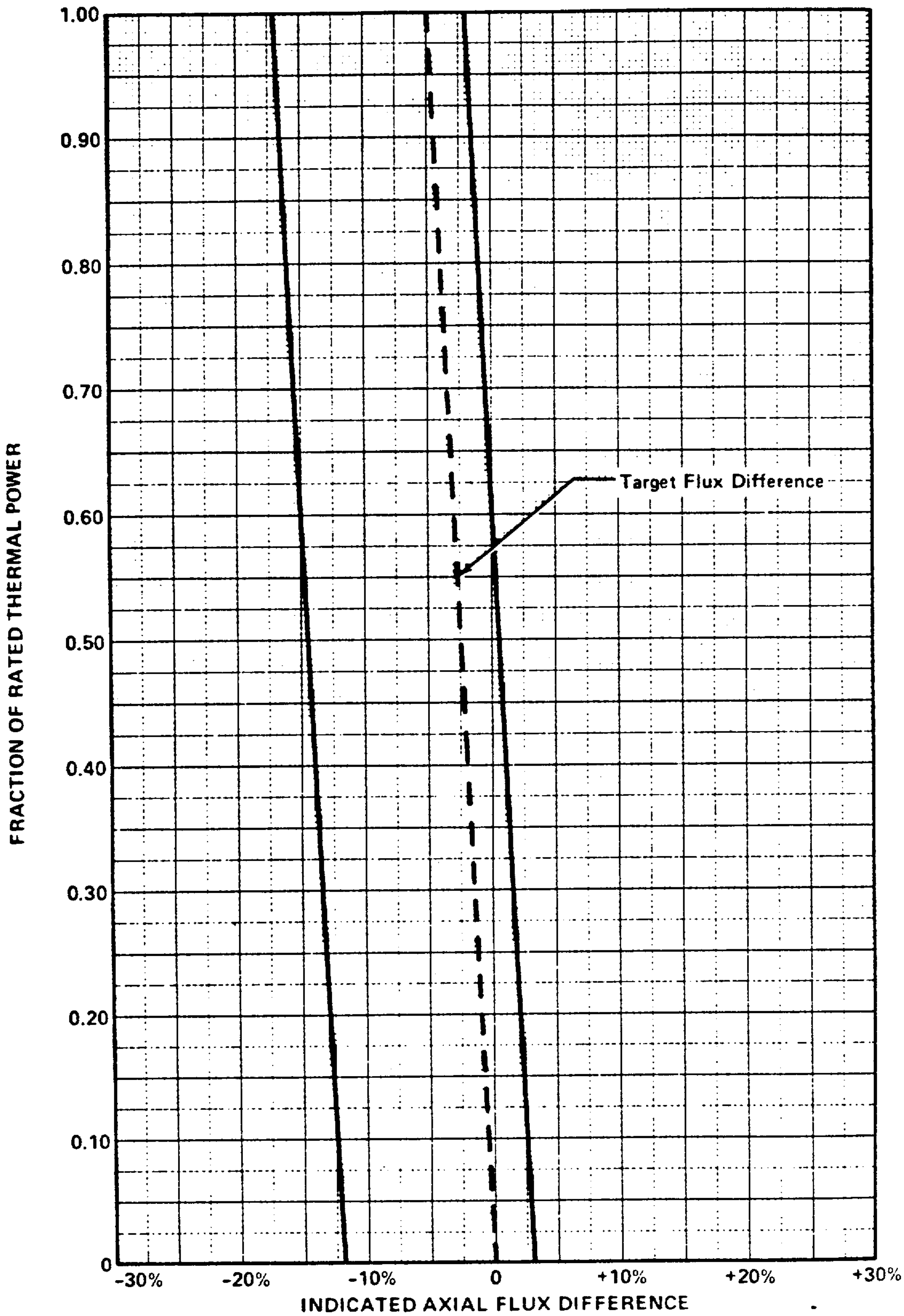

FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER (Unit 2)

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.

$F^N_{\Delta H}$ 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.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

For Unit 1 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.

BASES

HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

For Unit 2 the Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.9 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual 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 Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PCP and to the ODCM, pursuant to Specifications 6.13 and 6.14, respectively, as well as any major changes to Liquid, Gaseous or Solid Radwaste Treatment Systems, pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.10 or 3.3.3.11, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

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 Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.9 For Unit 1 the W(z) functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be provided to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 at least 60 days prior to cycle initial criticality. In the event that these values would be submitted at some other time during core life, it will be submitted 60 days prior to the date the values would become effective unless otherwise exempted by the Commission.

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.

For Unit 2 the F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided to the Regional Administrator of the Regional Office of the NRC with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, for all core planes containing bank "D" control rods and all unrodded core planes and the plot of predicted ($F_q^T \cdot P_{Rel}$) vs Axial Core Height with the limit envelope at least 60 days prior to each cycle initial criticality unless otherwise

RADIAL PEAKING FACTOR LIMIT REPORT (Continued)

approved by the Commission by letter. In addition, in the event that the limit should change requiring a new submittal or an amended submittal to the Radial Peaking Factor Limit Report, it will be submitted 60 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter. Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

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

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

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

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

c. All REPORTABLE EVENTS;

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

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

f. Records of radioactive shipments;

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

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1;

f. Records of reactor tests and experiments;

g. Records of training and qualification for current members of the unit staff;

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 14 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 6 TO FACILITY OPERATING LICENSE NPF-52

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DUKE POWER COMPANY, ET AL.

I. INTRODUCTION

By letter dated July 15, 1986 (Ref. 1), Duke Power Company (the licensee) made an application to amend facility operating licenses NPF-35 and NPF-52 for Catawba Nuclear Station Units 1 and 2, respectively, to reflect the Cycle 2 refueling and related Technical Specification (TS) changes for Unit 1 and a TS change for both Units related to application of a positive Moderator Temperature Coefficient (MTC). A second letter (Ref. 2) provided some TS pages inadvertently omitted from Reference 1. The proposed TS change applicable to both Units consists of increasing the allowable positive MTC and most negative EOL MTC. The previous TS allowed a +5 pcm/°F MTC at power levels up to 70% power, and 0 at power levels about 70%. The proposed revision would allow an MTC of +7 pcm/°F up to 70%, decreasing linearly above 70% power to 0 pcm/°F at 100% power.

II. EVALUATION

1. General Design

The Catawba Unit 1, 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 mechanical design of the Region 4 assemblies is the same as that of Regions 1, 2 and 3 except for the use of 304L stainless steel sleeves on the top grid, a small downward axial shift of the fuel rods and minor top grid modifications. The Region 4 fuel has been designed according to the fuel performance model in WCAP-8785 (Ref. 3). The fuel is designed and operated so that clad flattening will not occur as predicted by the Westinghouse model in WCAP-8377 (proprietary) and in WCAP-8381 (non-proprietary) (Ref. 4). For all fuel regions, the fuel rod internal pressure design basis, which is discussed and shown acceptable in WCAP-8964 (Ref. 5), is satisfied.

The licensee provided a Reload Safety Evaluation (RSE) for Catawba 1 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-9272-P-A (Ref. 6).

2. Nuclear Design

The Cycle 2 core loading is designed to meet an $[F_Q(Z) \times P]$ ECCS limit of $\leq 2.32 \times K(Z)$. Adherence to the F_Q limit is obtained by using the F_Q TS surveillance described in WCAP-10217-A (Ref. 7). 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, increased to account for expected plant maneuvers, to the 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 7 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 ΔI band, particularly at reduced power. RAOC is described in Reference 7 and was approved by the staff. Thus, it is acceptable for use in Catawba Unit 1.

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 TS to recognize this reduction in core peaking factors. The proposed Base Load TS are identical to those that the staff has previously approved for McGuire Units 1 and 2, 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 these results 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.

A more positive MTC than the current value is specified for Cycle 2. This is evaluated elsewhere in this SER.

3. <u>Thermal and Hydraulic Design</u>

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 Over-Temperature Delta-T setpoints. These generated a change to the $F(\Delta I)$ function in the TS. The change is acceptable because it results from cycle-specific calculations using approved methods (Refs. 6 and 7). Therefore, the staff concludes that the Cycle 2 thermal-hydraulic analysis is acceptable.

4. Accident Analysis

All the Cycle 2 kinetics parameters fall within the bounds upon which the previous applicable safety analysis is based, except for the proposed change to the positive MTC, and the following reanalysis.

The Uncontrolled Boron Dilution events for full power operation and startup operation were reanalyzed to show that there is greater than 15 minutes, from time of alarm for operator action to terminate the dilution before the minimum allowable shutdown margin is lost. The events were reanalyzed because the Cycle 1 analysis was cycle specific. The results show 156 minutes are available for full power operation with the reactor in automatic control and 65 minutes with the reactor at full power and in manual control. The latter result bounds the case for startup operation. Thus the results of the reanalysis are acceptable.

The licensee provided a report on the effect of the MTC change on accident analysis as an attachment to Reference 1. The analysis applies to both Catawba Units 1 and 2, and is evaluated below.

The licensee has assessed the impact of a positive MTC of 7 pcm/°F on the accident analyses presented in Chapter 15 of the FSAR. Those incidents which were found to be sensitive to positive or near-zero moderator coefficients were reanalyzed. These incidents are limited to transients which cause the reactor coolant temperature to increase. Accidents not reanalyzed included those resulting in excessive heat removal from the reactor coolant system, for which a large negative moderator coefficient is more limiting, and those for which heatup effects following reactor trip are not sensitive to the moderator coefficient. The staff agrees with the licensee's conclusions about which transients did and did not require reanalysis. The transients not reanalyzed are:

(1) RCCA misalignment/drop.
(2) Startup of an inactive reactor coolant loop.
(3) Excessive heat removal due to feedwater system malfunction.
(4) Excessive load increase.
(5) Spurious actuation of safety injection.
(6) Rupture of a main steam pipe.
(7) Loss-of-coolant accident (LOCA)

The incidents reanalyzed, with two exceptions, used a +7 pcm/°F moderator temperature coefficient, assumed to remain constant for variations in temperature. This is conservative, since the proposed change will require the coefficient to ramp to zero at full power. The two exceptions are the rod ejection and the rod withdrawal from subcritical accidents, for which the computer model cannot accept a constant coefficient. The coefficient decrease which occurred during the transients was less than the proposed change, which is acceptable. The transients reanalyzed and their results are:

A. Boron Dilution

Boron dilution accidents during refueling or startup are terminated by operator action. The proposed MTC does not reduce the time available for operator action in these modes below the acceptable value of 30 minutes from the time the operator is alerted to reactor criticality. This is acceptable. The dilution analysis for power conditions with the reactor in automatic control assumes operator action based on the rod insertion alarm. Analysis of the transient shows the time for operator action remains above the acceptable value of 15 minutes. The dilution from power with the reactor in manual control is bounded by the rod withdrawal transient. Boron dilution accident results will therefore remain acceptable with the proposed MTC.

B. Control Rod Bank Withdrawal from a Subcritical Condition

This transient results in an uncontrolled addition of reactivity leading to a power excursion causing a heatup of the moderator and fuel. The time the core is critical before a reactor trip is very short so that the RCS temperature does not increase significantly; hence the effect of a positive MTC is small. The analysis results show a transient average heat flux which does not exceed the steady state full power value and an increased core water temperature that remains below the full power value. The results show that the DNBR remains above the limit value during the transient, which is acceptable.

C. Uncontrolled Control Rod Bank Assembly Withdrawal at Power

This transient produces a mismatch in steam flow and core power, resulting in an increase in RCS temperature. However, the results show that the nuclear flux and overtemperature ΔT trips prevent the core minimum DNBR from falling below the limit value for this transient, which is acceptable.

D. Loss of Coolant Flow

The most severe loss of flow transient is caused by the simultaneous loss of power to all four reactor coolant pumps (RCPs). This case was reanalyzed to determine the effect of the positive MTC on the nuclear power transient and the resultant effect on the minimum DNBR reached during the transient. The minimum DNBR remains above the limit value during the transient, which is acceptable.

E. Locked Rotor

The locked event was reanalyzed because of the potential effect of the positive MTC on the nuclear power transient and thus on the RCS pressure and fuel temperature. A positive MTC will not affect the time to DNB because DNB is conservatively assumed to occur at the beginning of the transient. The results show peak RCS pressure and peak pellet average and peak cladding temperatures less than the limits used in the previously approved FSAR analyses, which is acceptable.

F. Loss of External Electric Load

The loss of external electric load transient was reanalyzed for both the beginning-of-life (BOL) and end-of-life cases. Since the MTC will be negative at end-of-life, the end-of-life results were essentially the same as in the FSAR. Two beginning-of-life cases were analyzed: (1) reactor in the automatic rod control mode with operation of the pressurizer spray and pressurizer power operated relief valves (PORV); and (2) reactor in the manual control mode with no credit for pressurizer spray or PORVs. The result of a loss of load is a core power that momentarily exceeds the secondary system power removal, causing an increase in RCS coolant temperature. The reactivity addition due to a positive MTC causes an increase in both nuclear power and RCS pressure. The result for the control rods in automatic control and assuming pressurizer spray and relief at BOL is a RCS pressure of 2518 psia following a reactor trip on high pressurizer pressure. A minimum DNBR well above the applicable limits is reached shortly after reactor trip. The result for the case of rods in manual control with no credit for pressure control is a peak RCS pressure of 2563 psia following a reactor trip on high pressure. The minimum DNBR increases throughout the transient. Since the DNBR remains above the applicable limits and the peak RCS pressure is less than 110% of the design value of 2500 psia, the conclusions presented in the previously approved FSAR analysis are still applicable.

G. Loss of Normal Feedwater/Loss of Offsite Power

These accidents are analyzed to show the ability of the secondary system auxiliary feedwater to remove decay heat from the reactor coolant system. The results show that the capacity of the auxiliary feedwater system is adequate to provide sufficient heat removal from the RCS. For the case without offsite power, the results verify that the natural circulation capacity of the RCS provide sufficient heat removal capability to prevent fuel or clad damage following reactor coolant pump coastdown.

H. Rupture of Main Feedwater Pipe

This accident is analyzed to demonstrate the ability of the secondary system auxiliary feedwater to remove heat from the RCS. The results show that the capacity of the auxiliary feedwater system is adequate to provide sufficient heat removal from the RCS to prevent overpressurization or core uncovery. For the case without offsite power, the results verify the natural circulation capacity of the RCS to prevent overpressurization and fuel or clad damage following reactor coolant pump coastdown.

I. Control Rod Ejection

The rod ejection transient was reanalyzed only for BOC since the MTC will be negative at EOC and the existing FSAR analysis remains applicable for EOC. The higher nuclear power levels and hotspot fuel temperatures resulting from a rod ejection are increased by a positive MTC. The results from the BOC reanalysis show that the fuel and clad temperatures are within the limiting values specified in the existing FSAR analysis. The peak hotspot fuel centerline temperature exceeded the melting temperature for the full power case; however, melting was restricted to less than the innermost ten percent of the pellet. The fuel and clad temperatures do not exceed the limits specified in the previously approved FSAR analysis. Therefore, the results of the control rod ejection reanalysis are acceptable.

J. Accidental Depressurization of the Reactor Coolant System

The acceptance criteria for the accidental depressurization of the RCS were shown to be satisfied by predicting a minimum DNBR above the limit value for this transient.

Since the reanalysis of the affected plant transients does not result in exceeding any of the fuel limits or safety limits specified in the previously approved reference or FSAR analyses, the staff concludes the analysis supporting operation with a positive moderator temperature coefficient of +7 pcm/°F up to 70% power, and decreasing linearly from this to 0 pcm/°F at full power will not pose an undue risk to the health and safety of the public and is therefore acceptable. The analysis is applicable to both Catawba Units 1 and 2, and therefore the proposed revision of the TS to incorporate the MTC for both Units is acceptable.

5. Technical Specification Changes

The TS Changes proposed in the licensee's submittals (Refs. 1 and 2) involves the following changes for Catawba Unit 1 only:

1. RAOC and Axial Flux Difference Limits
2. F_Q Surveillance
3. Base Load Technical Specifications
4. Rod Insertion Limits
5. OT $\Delta Tf(\Delta I)$

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 1 only but the actual change pages involve both Units, making the changes for Unit 1 and leaving the Unit 2 TS unchanged. The revisions to the bases are also acceptable.

In addition, the positive moderator coefficient change for both Units was found acceptable in Section 4, Accident Analysis. A second change to the moderator coefficient revised the most negative EOC coefficient to the value used in the accident analysis for both Units and is, therefore, acceptable.

III. ENVIRONMENTAL CONSIDERATION

The amendments involve a change in use of facility components located within the restricted area as defined in 10 CFR Part 20 and a change in surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there have been no public comments 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 enviromental assessment need be prepared in connection with the issuance of the amendments.

IV. CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (51 FR 30567) on August 27, 1986, and 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 to H. R. Denton (NRC) from H. B. Tucker (Duke Power), "Catawba Nuclear Station, Docket Nos. 50-413 and 50-414 Catawba 1/Cycle 2 Reload," July 15, 1986.

(2) Letter to H. R. Denton (NRC) from H. B. Tucker (Duke Power), "Catawba Nuclear Station, Docket No. 50-413, Catawba Unit 1 Cycle 2 Reload," July 24, 1986.

(3) Miller, J.V., (Ed.), "Improved Analytical Model Used in Westinghouse Fuel Rod Design Computations," WCAP-8785, October 1976.

(4) George, R.A., (et. al.), "Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381 (Non-Proprietary) July 1974.

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

(6) Davidson, S.L., et. al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

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

Principal Contributors: Kahtan Jabbour, PWR#4/DPWR-A
Marvin Dunenfeld, RSB/DPWR-A

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