May 31, 1991

Docket Nos. 50-413
and 50-414

Distribution
See next page

Mr. M.S. Tuckman
Vice President -
Nuclear Operations
Duke Power Company
P.O. Box 1007
Charlotte, North Carolina 28201-1007

Dear Mr. Tuckman:

SUBJECT: ISSUANCE OF AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NPF-35 AND AMENDMENT NO. 80 TO FACILITY OPERATING LICENSE NPF-52 - CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (TACS)

SUBJECT: ISSUANCE OF AMENDMENT NO. TO FACILITY OPERATING LICENSE NPF-35 AND AMENDMENT NO. TO FACILITY OPERATING LICENSE NPF-52 - CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (TACS 79419/79420)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. to Facility Operating License NPF-35 and Amendment No. to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated January 9, 1991.

The amendments revise the TSs to reflect fuel reloading for Catawba Unit 1's Cycle 6 operation with fuel manufactured by the B&W Fuel Company and to continue to reflect the previously existing TSs for Unit 2 fuel.

A copy of the related Safety Evaluation is also enclosed. Notice of issuance of the amendments will be included in the Commission's biweekly Federal Register notice.

Sincerely,
/S/

Robert E. Martin, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 86 to NPF-35
2. Amendment No. 80 to NPF-52
3. Safety Evaluation

cc w/enclosures:
See next page

OFFICIAL RECORD COPY
Document Name: RELOAD TS AMEND

* See attached.

LA:PDII3	PM:PDII3	OGC *	D:PDII3
RIngram	RMartin/rst		DMatthews
5/31/91	5/31/91	5/28/91	5/31/91

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

May 31, 1991

Docket Nos. 50-413
and 50-414

Mr. M.S. Tuckman
Vice President -
Nuclear Operations
Duke Power Company
P.O. Box 1007
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cc w/enclosures:
See next page

Mr. M.S. Tuckman
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. 86
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 Janaury 9, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations 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.

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. 80
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 January 9, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations 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 attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-52 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 80 , 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

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes

Date of Issuance: May 31, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 86

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO. 80

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 pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
III	III
IV	IV
V	Va
--	Vb
VII	VII*
VIII	VIII
XIII	XIIIa
--	XIIIb
2-1	2-1
2-2	2-2
--	2-2a
2-4	2-4
2-7	2-7
2-8	2-8
2-10	2-10
B 2-1 through B 2-2	B 2-1 through B 2-2a
3/4 2-1 through 3/4 2-16	3/4 A2-1 through 3/4 A2-16 and 3/4 B2-1 through 3/4 B2-16
3/4 5-1 thru 3/4 5-4a	3/4 5-1 thru 3/4 5-4
3/4 5-7	3/4 5-7*
3/4 5-8	3/4 5-8

B 3/4 2-1 thru B 3/4 2-6	B 3/4 2-1 thru B 3/4 2-10
6-19	6-19
6-19a	6-19a

*Overleaf pages

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.4 REACTOR COOLANT SYSTEM

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1a (Unit 1) and 2.1-1b (Unit 2) for four loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

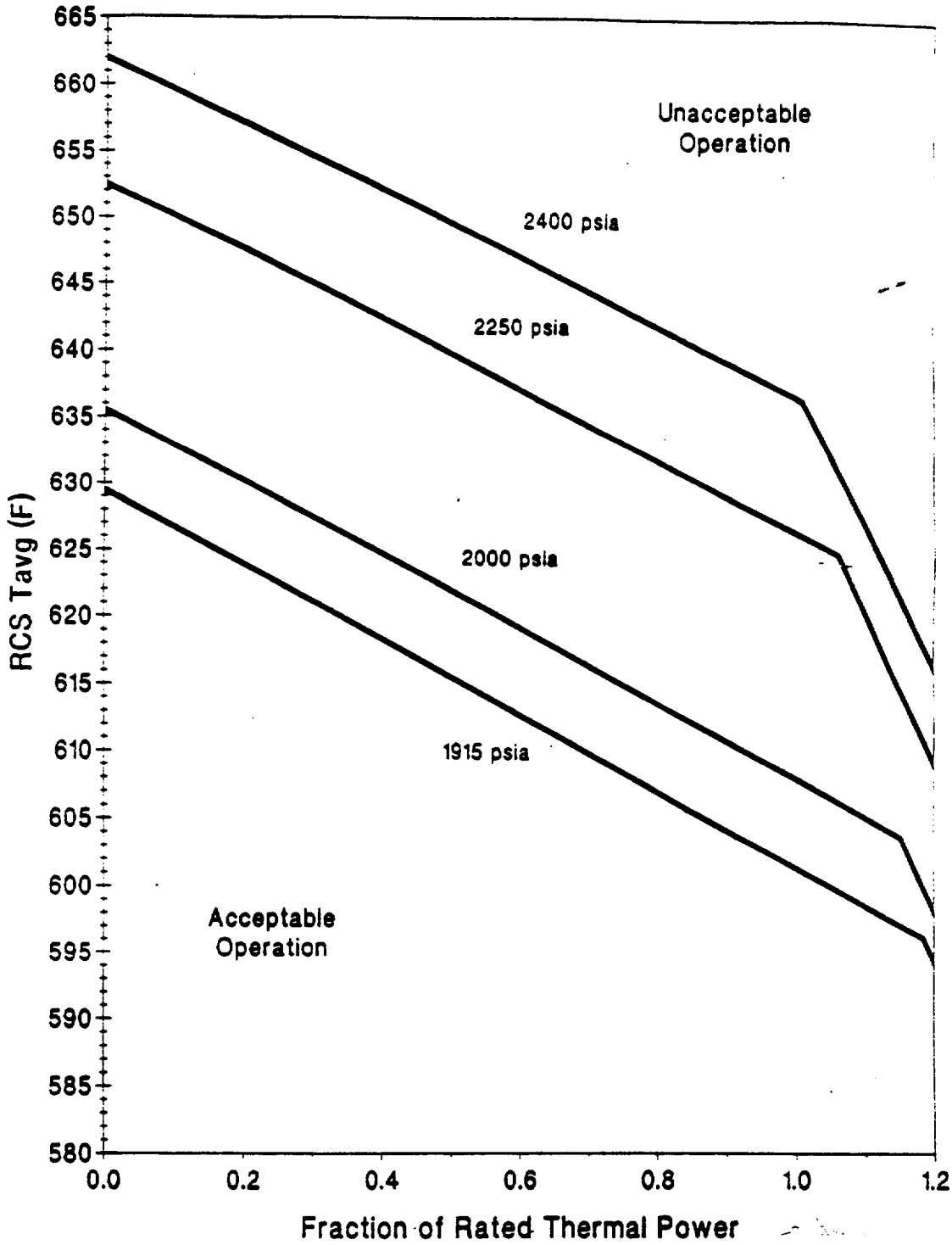

FIGURE 2.1-1a

REACTOR CORE SAFETY LIMITS - FOUR LOOPS IN OPERATION, UNIT 1

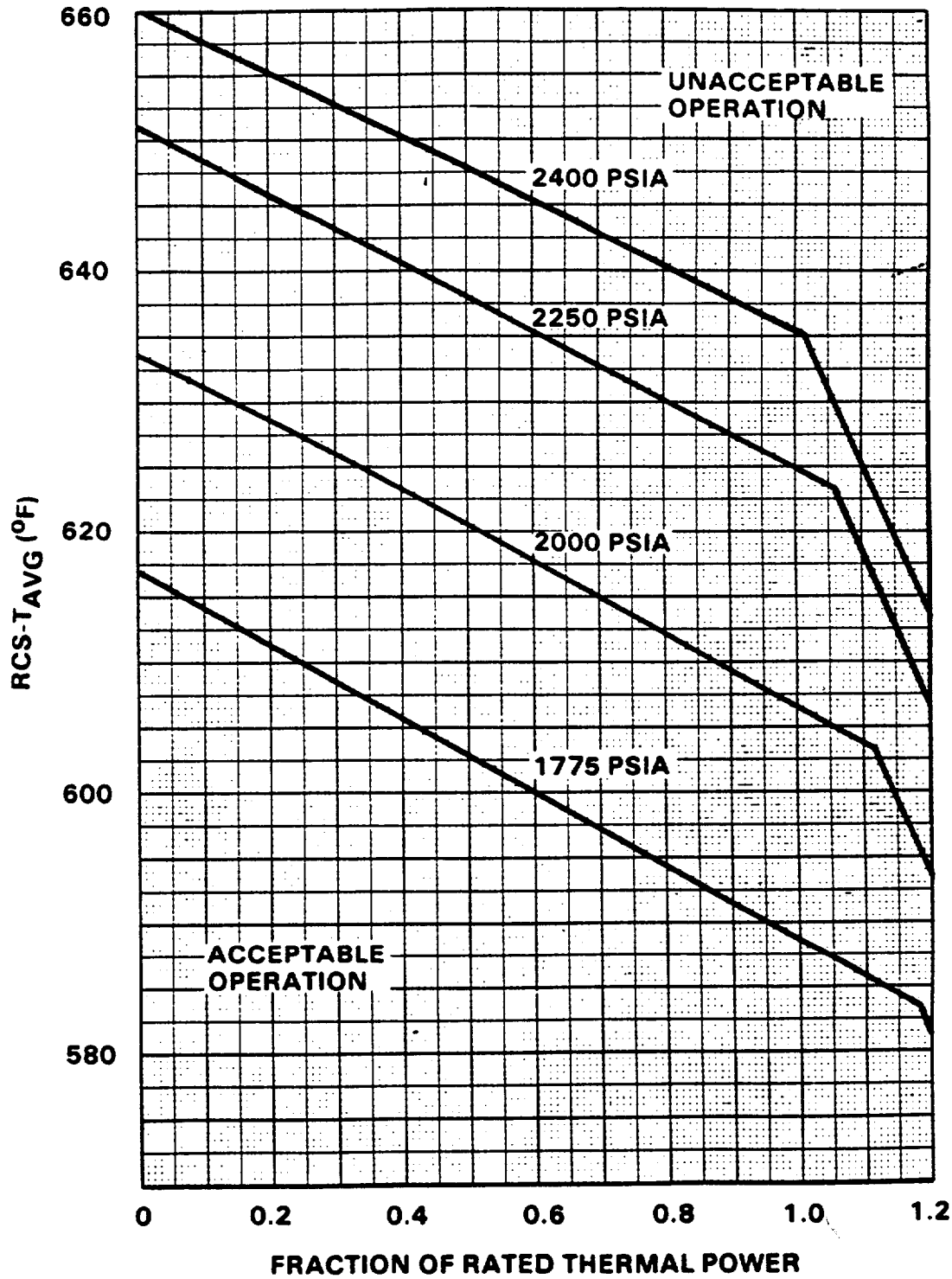

FIGURE 2.1-1b

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION, UNIT 2

TABLE 2.2.-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	5.92	0	≤109% of RTP*	≤110.9% of RTP*
b. Low Setpoint	8.3	5.92	0	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥ 2 seconds	≤6.3% of RTP* with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP*	≤31% of RTP*
6. Source Range, Neutron Flux	17.0	10	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT	6.98	3.0	2.12	See Note 1	See Note 2
8. Overpower ΔT	4.9	1.24	1.7	See Note 3	See Note 4
9. Pressurizer Pressure-Low	4.0	2.21	1.5	≥1945 psig	≥1938 psig***
10. Pressurizer Pressure-High	7.5	0.71	0.5	≤2385 psig	≤2399 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span
12. Reactor Coolant Flow-Low	2.92	1.48	0.6	≥90% of loop minimum measured flow**	≥88.9% of loop minimum measured flow**

*RTP = RATED THERMAL POWER

**Loop minimum measured flow = 96,900 gpm (Unit 2), 96,250 gpm (Unit 1)

***Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to these values.

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by Loop Narrow Range RTDs;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 12$ s, $\tau_2 = 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.38;
K_2	=	0.02401/°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}, $\tau_4 = 22$ s, $\tau_5 = 4$ s;
T	=	Average temperature, °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, $\tau_6 = 0$;

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%, $f_1(\Delta I) = 0$, 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 3.0%.

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

NOTE 3: (Continued)

K_6 = 0.001707/°F for T > 590.8°F and K_6 = 0 for T $\leq$ 590.8°F,

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 590.8°F),

S = As defined in Note 1, and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.8%.

2.1.1 REACTOR CORE (FOR UNIT 1)

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

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

The DNB design basis is as follows: there must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the BWCMV correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95% probability with 95% confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and in the BWCMV DNB correlation are considered statistically such that there is at least a 95% confidence that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty is used to establish a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

These curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.49 for Westinghouse Optimized Fuel Assemblies (OFA's) and 1.55 for the BWFC Mark-BW Fuel Assemblies and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)] \text{ For the Westinghouse OFA's}$$

$$F^N_{\Delta H} = 1.55\ [1 + 0.3\ (1-P)] \text{ For the BWFC Mark-BW's}$$

Where P is the fraction of RATED THERMAL POWER.

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

2.1.1 REACTOR CORE (FOR UNIT 2)

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

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the WRB-1 correlation. The WRB-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio, (DNBR), is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95% probability with 95% confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% confidence that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

This curve is based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

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

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the Reactor Coolant System piping, valves, and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire Reactor Coolant System is hydrotested at 125% (3110 psig) of design pressure, to demonstrate integrity prior to initial operation.

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 the acceptable limits specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER.* (Unit 1)

ACTION:

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

   1. Either restore the indicated AFD to within the COLR 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. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

*See Special Test Exceptions Specification 3.10.2.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

c. The provisions of Specification 4.0.4 are not applicable.

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

POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(X,Y,Z)$ shall be limited by imposing the following relationships:

$$F_Q^{MA}(X,Y,Z) \leq \frac{F_Q^{RTP}}{P} K(Z) \text{ for } P > 0.5$$

$$F_Q^{MA}(X,Y,Z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) \text{ for } P \leq 0.5$$

Where: F_Q^{RTP} = the F_Q Limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

$F_Q^{MA}(X,Y,Z)$ = the measured heat flux hot channel factor $F_Q^{M}(X,Y,Z)$, with adjustments as specified in 4.2.2.3,

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

$K(Z)$ = the normalized $F_Q(X,Y,Z)$ limit specified in the COLR for the appropriate fuel types.

APPLICABILITY: MODE 1. (Unit 1)

ACTION:

With $F_Q(X,Y,Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours, and

b. Control the AFD to within new AFD limits which are determined by reducing the allowable power at each point along the AFD limit lines of Specification 3.2.1 at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 15 minutes and reset the AFD alarm setpoints to the modified limits within 8 hours, and

c. 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% $F_Q^{MA}(X,Y,Z)$ exceeds the limit, and

d. 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 $F_Q(X,Y,Z)$ is demonstrated through incore mapping to be within its limit.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q^M(X,Y,Z)$[1] shall be evaluated to determine whether $F_Q(X,Y,Z)$ 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. Measuring $F_Q^M(X,Y,Z)$ at the earliest of:

1. At least once per 31 Effective Full Power Days, or

2. Upon reaching equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q^M(X,Y,Z)$ was last determined[2], or

3. At each time the QUADRANT POWER TILT RATIO indicated by the excore detectors is normalized using incore detector measurements.

c. Performing the following calculations:

1. For each location, calculate the % margin to the maximum allowable design as follows:

$$\% \text{ Operational Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{[F_Q^L(X,Y,Z)]^{OP}}\right) \times 100\%$$

$$\% \text{ RPS Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{[F_Q^L(X,Y,Z)]^{RPS}}\right) \times 100\%$$

where $[F_Q^L(X,Y,Z)]^{OP}$ and $[F_Q^L(X,Y,Z)]^{RPS}$ are the Operational and RPS design peaking limits defined in the COLR.

2. Find the minimum Operational Margin of all locations examined in 4.2.2.2.c.1 above. If any margin is less than zero, then either of the following actions shall be taken:

---

(1) No additional uncertainties are required in the following equations for $F_Q^M(X,Y,Z)$, because the limits include uncertainties.

(2) During power escalation at the beginning of each cycle, THERMAL POWER may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

(a) Within 15 minutes:

(1) Control the AFD to within new AFD limits that are determined by:

$$(\text{AFD Limit})^{\text{reduced}}_{\text{negative}} = (\text{AFD Limit})^{\text{COLR}^{(3)}}_{\text{negative}}$$

$$+ [\text{NSLOPE}_1{}^{(3)} \times \text{Margin}^{\text{min}}_{\text{OP}}] \text{ absolute value}$$

$$(\text{AFD Limit})^{\text{reduced}}_{\text{positive}} = (\text{AFD Limit})^{\text{COLR}^{(3)}}_{\text{positive}}$$

$$-[\text{PSLOPE}_1{}^{(3)} \times \text{Margin}^{\text{min}}_{\text{OP}}] \text{ absolute value}$$

where $\text{Margin}^{\text{min}}_{\text{OP}}$ is the minimum margin from 4.2.2.2.c.1, and

(2) Within 8 hours, reset the AFD alarm setpoints to the modified limits of 4.2.2.2.c.2.a, or

(b) Comply with the ACTION requirements of Specification 3.2.2.

3. Find the minimum RPS Margin of all locations examined in 4.2.2.2.c.1 above. If any margin is less than zero, then the following action shall be taken:

Within 72 hours, reduce the K_1 value for OTΔT by:

$$K_1 \text{ adjusted} = K_1{}^{(4)} - [\text{KSLOPE}^{(3)} \times \text{Margin}^{\text{min}}_{\text{RPS}}] \text{absolute value}$$

where $\text{MARGIN}^{\text{min}}_{\text{RPS}}$ is the minimum margin from 4.2.2.2.c.1.

(3) Defined and specified in the COLR per Specification 6.9.1.9.

(4) K_1 value from Table 2.2-1.

d. Extrapolating the two most recent measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

$$[F_Q^M(X,Y,Z] \text{ (extrapolated)} \geq [F_Q^L(X,Y,Z]^{OP} \text{ (extrapolated), or}$$

$$[F_Q^M(X,Y,Z] \text{ (extrapolated)} \geq [F_Q^L(X,Y,Z]^{RPS} \text{ (extrapolated),}$$

either of the following actions shall be taken:

1. $F_Q^M(X,Y,Z)$ shall be increased by 2 percent over that specified in 4.2.2.2.a, and the calculations of 4.2.2.2.c repeated, or

2. A movable incore detector power distribution map shall be obtained, and the calculations of 4.2.2.2.c.1 shall be performed no later than the time at which the margin in 4.2.2.2.c.1 is extrapolated to be equal to zero.

e. The limits in Specifications 4.2.2.2.c and 4.2.2.2.d are not applicable in the following core plane 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.

4.2.2.3 When a full core power distribution map is taken for reasons other than meeting the requirements of Specification 4.2.2.2, an overall $F_Q^M(X,Y,Z)$ shall be determined, then increased by 3% to account for manufacturing tolerances, further increased by 5% to account for measurement uncertainty, and further increased by the radial-local peaking factor to obtain a maximum local peak. This value shall be compared to the limit in Specification 3.2.2.

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

3.2.3 $F_{\Delta H}(X,Y)$ shall be limited by imposing the following relationship:

$$F_{\Delta H}R^M(X,Y) \leq F_{\Delta H}R^L(X,Y)$$

Where: $F_{\Delta H}R^M(X,Y)$ = the maximum measured radial peak ratio as defined in the CORE OPERATING LIMITS REPORT (COLR).

$F_{\Delta H}R^L(X,Y)$ = the maximum allowable radial peak ratio as defined in the (COLR).

APPLICABILITY: MODE 1. (UNIT 1)

ACTION:

With $F_{\Delta H}(X,Y)$ exceeding its limit:

a. Within 2 hours, reduce the allowable THERMAL POWER from RATED THERMAL POWER at least RRH%[1] for each 1% that $F_{\Delta H}R^M(X,Y)$ exceeds the limit, and

b. Within 6 hours either:

   1. Restore $F_{\Delta H}R^M(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

   2. Reduce the Power Range Neutron Flux-High Trip Setpoint in Table 2.2-1 at least RRH% for each 1% that $F_{\Delta H}R^M(X,Y)$ exceeds that limit, and

c. Within 72 hours of initially being outside the limit of Specification 3.2.3, either:

   1. Restore $F_{\Delta H}R^M(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

   2. Perform the following actions:

      (a) Reduce the OTΔT K_1 term in Table 2.2-1 by at least TRH[2] for each 1% that $F_{\Delta H}R^M(X,Y)$ exceeds the limit, and

      (b) Verify through incore mapping that $F_{\Delta H}R^M(X,Y)$ is restored to within the limit for the reduced THERMAL POWER allowed by ACTION a, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

---

[1] RRH is the amount of THERMAL POWER reduction required to compensate for each 1% that $F_{\Delta H}R^M(X,Y)$ exceeds $F_{\Delta H}R^L(X,Y)$ provided in the COLR per Specification 6.9.1.9.

[2] TRH is the amount of OTΔT K_1 setpoint reduction required to compensate for each 1% that $F_{\Delta H}R^M(X,Y)$ exceeds the limit of Specification 3.2.3, provided in the COLR per Specification 6.9.1.9.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

d. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a. and/or c.2., above; subsequent POWER OPERATION may proceed provided that $F_{\Delta H}R^M(X,Y)$ is demonstrated, through incore flux mapping, to be within the limit specified in the COLR prior to exceeding the following THERMAL POWER levels:

1) 50% of RATED THERMAL POWER,

2) 75% of RATED THERMAL POWER, and

3) Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F_{\Delta H}R^M(X,Y)$ shall be evaluated to determine whether $F_{\Delta H}(X,Y)$ 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. Measuring $F_{\Delta H}R^M(X,Y)$ according to the following schedule:

1. Prior to operation above 75% of RATED THERMAL POWER at the beginning of each fuel cycle, and the earlier of:

2. At least once per 31 Effective Full Power Days, or

3. At each time the QUADRANT POWER TILT RATIO indicated by the excore detectors is normalized using incore detector measurements.

c. Performing the following calculations:

1. For each location, calculate the % margin to the maximum allowable design as follows:

$$\%F_{\Delta H}\ \text{Margin} = 1 - \frac{F_{\Delta H}R^M(X,Y)}{F_{\Delta H}R^L(X,Y)} \times 100\%$$

No additional uncertainities are required for $F_{\Delta H}R^M(X,Y)$, because $F_{\Delta H}R^L(X,Y)$, includes uncertainties.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. Find the minimum margin of all locations examined in 4.2.3.2.c.1 above. If any margin is less than zero, comply with the ACTION requirements of Specification 3.2.3.

d. Extrapolating the two most recent measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

$F\Delta HR^{M}$ (extrapolated) $\geq$ $F\Delta HR^{L}$ (extrapolated)

either of the following actions shall be taken:

1. $F\Delta HR^{M}(X,Y)$ shall be increased by 2 percent over that specified in 4.2.3.2.a, and the calculations of 4.2.3.2.c repeated, or

2. A movable incore detector power distribution map shall be obtained, and the calculations of 4.2.3.2.c shall be performed no later than the time at which the margin in 4.2.3.2.c is extrapolated to be equal to zero.

POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER (Unit 1).*,**

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Within 2 hours either:

a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours 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; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

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*See Special Test Exceptions Specification 3.10.2.
**Not applicable until calibration of the excore detectors is completed subsequent to refueling.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02, within 30 minutes;

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours 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; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours 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; and

3. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg},

b. Pressurizer Pressure,

c. Reactor Coolant System Total Flow Rate.

APPLICABILITY: MODE 1. (Unit 1)

ACTION:

a. With either of the parameters identified in 3.2.5a. and b. above exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

b. With the combination of Reactor Coolant System total flow rate and THERMAL POWER within the region of restricted operation specified on Figure 3.2-1, within 6 hours reduce the Power Range Neutron Flux-High Trip Setpoint to below the nominal setpoint by the same amount (% RTP) as the power reduction required by Figure 3.2-1.

c. With the combination of Reactor Coolant System total flow rate and THERMAL POWER within the region of prohibited operation specified on Figure 3.2-1:

1. Within 2 hours either:

a) Restore the combination of Reactor Coolant System total flow rate and THERMAL POWER to within the region of permissible operation, or

b) Restore the combination of Reactor Coolant System total flow rate and THERMAL POWER to within the region of restricted operation and comply with action a. above, or

c) Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

2. Within 24 hours of initially being within the region of prohibited operation specified on Figure 3.2-1, verify that the combination of THERMAL POWER and Reactor Coolant System total flow rate are restored to within the regions of restricted or permissible operation, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

4.2.5.3 The Reactor Coolant System total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER		LIMITS Four Loops in Operation
Average Temperature		
Meter Average	- 4 channels:	≤ 592°F
	- 3 channels:	≤ 592°F
Computer Average	- 4 channels:	≤ 593°F
	- 3 channels:	≤ 593°F
Pressurizer Pressure		
Meter Average	- 4 channels:	≥ 2227 psig*
	- 3 channels:	≥ 2230 psig*
Computer Average	- 4 channels:	≥ 2222 psig*
	- 3 channels:	≥ 2224 psig*
Reactor Coolant System Total Flow Rate		Figure 3.2-1

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

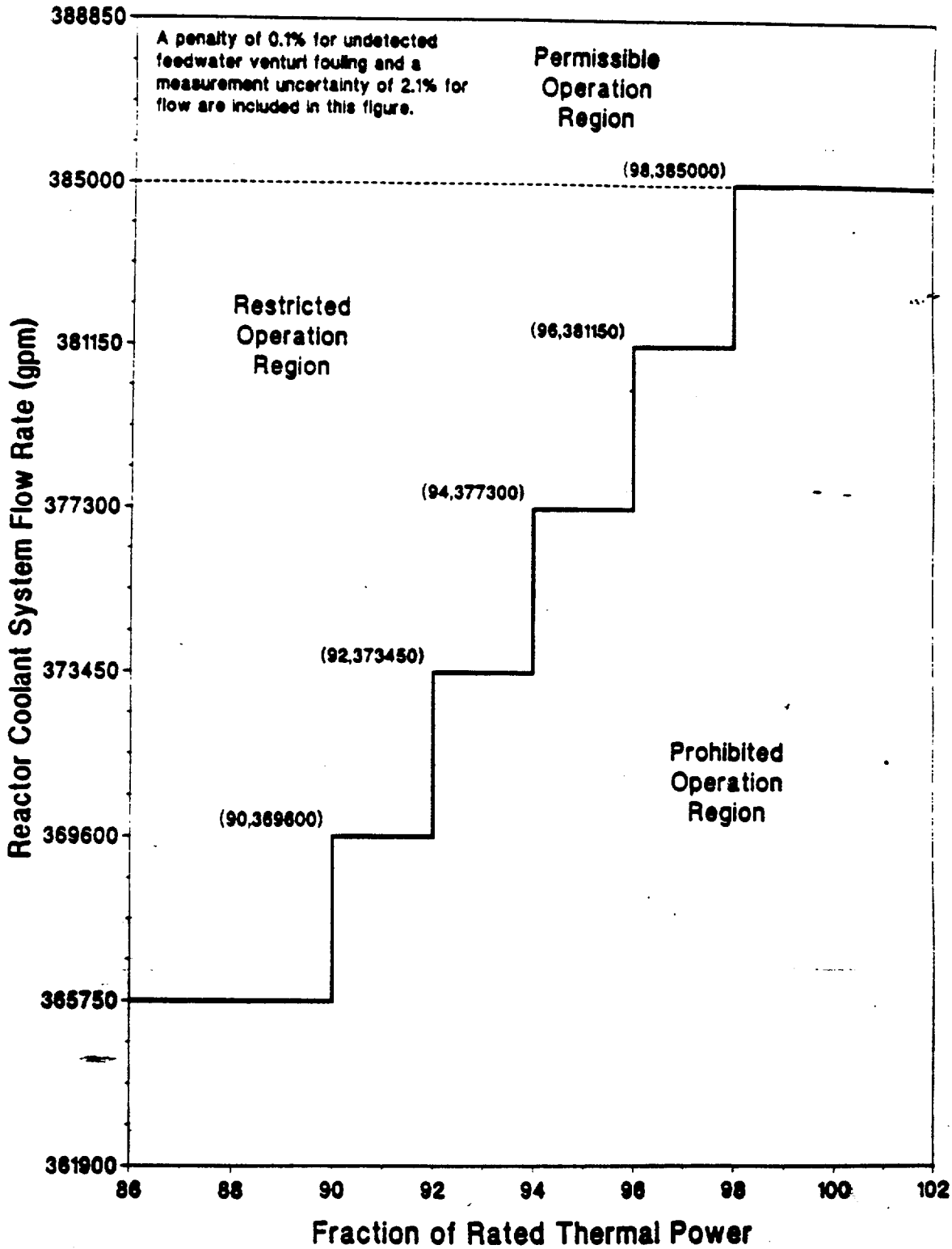

Figure 3.2-1 Reactor Coolant System Total Flow Rate Versus Rated Thermal Power - Four Loops in Operation (Unit 1)

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 as specified in the CORE OPERATING LIMITS REPORT (COLR) for RAOC operation, or

b. within the target band specified in the COLR about the target flux difference during baseload operation.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER* (Unit 2)

ACTION:

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

   1. Either restore the indicated AFD to within the COLR 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 COLR specified target band limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue Base Load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

---

*See Special Test Exceptions Specification 3.10.2.

**APL^{ND} is the minimum allowable (nuclear design) power level for base load operation and is specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

c. The provisions of Specification 4.0.4 are not applicable.

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

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

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

POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

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

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

Where: F_Q^{RTP} = the F_Q Limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

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

K(Z) = the normalized $F_Q(Z)$ for a given core height specified in the COLR.

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.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

c. Satisfying the following relationship:

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

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

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(z), and W(z) are specified in the CORE OPERATING LIMITS REPORT 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.

POWER DISTRIBUTION LIMITS

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:

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

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

2) One of the following actions shall be taken:

a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits of Specification 3.2.1 by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

c) Verify that the requirements of Specification 4.2.2.3 for Base Load operation are satisfied and enter Base Load operation.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

g. The limits specified in Specifications 4.2.2.2c., 4.2.2.2e., and 4.2.2.2f., above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive
2. Upper core region from 85 to 100%, inclusive.

4.2.2.3 Base Load operation is permitted at powers above APL^{ND*} if the following conditions are satisfied:

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within the target band about the target flux difference of Specification 3.2.1) 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:

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

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(Z)$ as a function of core height. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during Base Load operation. F_Q^{RTP}, K(z), and $W(Z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT 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.

*APL^{ND} is the minimum allowable (nuclear design) power level for Base Load operation in Specification 3.2.1.

SURVEILLANCE REQUIREMENTS (Continued)

c. Satisfying the following relationship:

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

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. F_Q^{RTP} is the F_Q limit. $K(Z)$ is the normalized $F_Q(Z)$ as a function of core height. P is the relative THERMAL POWER. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during Base Load operation. F_Q^{RTP}, $K(Z)$, and $W(Z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT 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

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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.

POWER DISTRIBUTION LIMITS

3/4.2.3 REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System total flow rate and R shall be maintained within the region of permissible operation specified in the CORE OPERATING LIMITS REPORT (COLR) for four loop operation.

Where:

a. $$R = \frac{F^N_{\Delta H}}{F^{RTP}_{\Delta H}[1.0 + MF_{\Delta H}(1.0 - P)]},$$

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

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since the figure specified in the COLR includes penalties for undetected feedwater venturi fouling of 0.1% and for measurement uncertainties of 2.1% for flow and 4% for incore measurement of $F^N_{\Delta H}$,

d. $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the COLR, and

e. $MF_{\Delta H}$ = The power factor multiplier specified in the COLR.

APPLICABILITY: MODE 1 (UNIT 2).

ACTION:

a. With the combination of Reactor Coolant System total flow rate and R within the region of restricted operation within 6 hours reduce the Power Range Neutron Flux-High Trip Setpoint to below the nominal setpoint by the same amount (% RTP) as the power reduction required by the figure specified in the COLR.

b. With the combination of Reactor Coolant System total flow rate and R within the region of prohibited operation specified in the COLR:

1. Within 2 hours either:

a) Restore the combination of Reactor Coolant System total flow rate and R to within the region of permissible operation, or

b) Restore the combination of Reactor Coolant System total flow rate and R to within the region of restricted operation and comply with action a. above, or

POWER DISTRIBUTION LIMITS

3/4.2.3 REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c) Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

2. Within 24 hours of initially being within the region of prohibited operation specified in the COLR, verify through incore flux mapping and Reactor Coolant System total flow rate comparison that the combination of R and Reactor Coolant System total flow rate are restored to within the regions of restricted or permissible operation, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

3. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION b.1.c) and/or b.2., above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated Reactor Coolant System total flow rate are demonstrated, through incore flux mapping and Reactor Coolant System total flow rate comparison, to be within the regions of restricted or permissible operation specified in the COLR prior to exceeding the following THERMAL POWER levels:

   a) A nominal 50% of RATED THERMAL POWER,

   b) A nominal 75% of RATED THERMAL POWER, and

   c) Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated Reactor Coolant System total flow rate determined by process computer readings or digital voltmeter measurement and R shall be determined to be within the regions of restricted or permissible operation specified in the COLR:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated Reactor Coolant System total flow rate shall be verified to be within the regions of restricted or permissible operation specified in the COLR at least once per 12 hours when the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The Reactor Coolant System total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

4.2.3.5 The Reactor Coolant System total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, *above 50% of RATED THERMAL POWER (Unit 2)

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

   a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

   b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Within 2 hours either:

   a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

   b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours 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; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

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*See Special Test Exceptions Specification 3.10.2.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1, within 30 minutes;

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours 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; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours 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; and

3. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}, and

b. Pressurizer Pressure.

APPLICABILITY: MODE 1, Unit 2 only.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER		LIMITS Four Loops in Operation
Average Temperature		
Meter Average	- 4 channels:	≤ 592°F
	- 3 channels:	≤ 592°F
Computer Average	- 4 channels:	≤ 593°F
	- 3 channels:	≤ 593°F
Pressurizer Pressure		
Meter Average	- 4 channels:	≥ 2227 psig*
	- 3 channels:	≥ 2230 psig*
Computer Average	- 4 channels:	≥ 2222 psig*
	- 3 channels:	≥ 2224 psig*

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The discharge isolation valve open,

b. A contained borated water volume of between 7704 and 8004 gallons,

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between 585 and 678 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve or boron concentration less than 1900 ppm, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one accumulator inoperable due to boron concentration less than 1900 ppm and:

   1) The volume weighted average boron concentration of the three limiting accumulators 1900 ppm or greater, restore the inoperable accumulator to OPERABLE status within 24 hours of the low boron determination or be in at least HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

   2) The volume weighted average boron concentration of the three limiting accumulators less than 1900 ppm but greater than 1500 ppm, restore the inoperable accumulator to OPERABLE status or return the volume weighted average boron concentration of the three limiting accumulators to greater than 1900 ppm and

*Reactor Coolant System pressure above 1000 psig.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

enter ACTION c.1 within 6 hours of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

3) The volume weighted average boron concentration of the three limiting accumulators 1500 ppm or less, return the volume weighted average boron concentration of the three limiting accumulators to greater than 1500 ppm and enter ACTION c.2 within 1 hour of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each cold leg injection accumulator isolation valve is open.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 75 gallons by verifying the boron concentration of the accumulator solution;

c. At least once per 31 days when the Reactor Coolant System pressure is above 2000 psig by verifying that power is removed from the isolation valve operators on Valves NI54A, NI65B, NI76A, and NI88B and that the respective circuit breakers are padlocked; and

d. At least once per 18 months by verifying that each cold leg injection accumulator isolation valve opens automatically under each of the following conditions:**

1) When an actual or a simulated Reactor Coolant System pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and

2) Upon receipt of a Safety Injection test signal.

** This surveillance need not be performed until prior to entering HOT STANDBY following the Unit 1 refueling.

SURVEILLANCE REQUIREMENTS (Continued)

4.5.1 Each cold leg injection accumulator water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

Page 3/4 5-4 is intentionally blank.

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 660 psig the interlocks will cause the valves to automatically close.

2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:**

1) Verifying that each automatic valve in the flow path actuates to its correct position on Safety Injection and Containment Sump Recirculation test signals, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) Residual heat removal pump.

f. By verifying that each of the following pumps develops the indicated differential pressure when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump $\geq$ 2380 psid,

2) Safety Injection pump $\geq$ 1430 psid, and

3) Residual heat removal pump $\geq$ 165 psid.

g. By verifying the correct position of each electrical and/or mechanical stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2) At least once per 18 months.

Centrifugal Charging Pump Injection Throttle Valve Number	Safety Injection Throttle Valve Number
NI-14	NI-164
NI-16	NI-166
NI-18	NI-168
NI-20	NI-170

** This surveillance need not be performed until prior to entering HOT SHUTDOWN following the Unit One first refueling.

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 345 gpm, and

b) The total pump flow rate is less than or equal to 565 gpm.

2) For Safety Injection pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 450 gpm, and

b) The total pump flow rate is less than or equal to 660 gpm.

3) For residual heat removal pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3648 gpm.

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 are not exceeded.

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

$F_Q(X,Y,Z)$ 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;

$F^N_{\Delta H}(X,Y)$ 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.

K(z) is defined as the normalized $F_Q(X,Y,Z)$ limit for a given core height.

3/4.2.1 AXIAL FLUX DIFFERENCE-Unit 1

The limits on AXIAL FLUX DIFFERENCE (AFD) specified in the CORE OPERATING LIMITS REPORT (COLR) ensure that the $F_Q(X,Y,Z)$ and the $F_{\Delta H}(X,Y)$ limits are not exceeded during either normal operation or in the event of xenon redistribution following power changes. The AFD envelope specified in the COLR has been adjusted for measurement uncertainty.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Unit 1)

The limits on heat flux hot channel factor, 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 ECCS acceptance criteria are not exceeded. The peaking limits are specified in the CORE OPERATING LIMITS REPORT (COLR) per Specification 6.9.1.9.

The heat flux hot channel factor and nuclear enthalpy rise hot channel factor are each 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:

HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Unit 1) (Continued)

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;

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

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

$F\Delta H(X,Y)$ will be maintained within its limits provided Conditions a. through d. above are maintained.

The limits on the nuclear enthalpy rise hot channel factor, $F\Delta H(X,Y)$, are specified in the COLR as Maximum Allowable Radial Peaking limits, obtained by dividing the Maximum Allowable Total Peaking (MAP) limit by the axial peak [AXIAL(X,Y)] for location (X,Y). By definition, the Maximum Allowable Radial Peaking limits will, for Mark-BW fuel, result in a DNBR for the limiting transient that is equivalent to the DNBR calculated with a design $F\Delta H(X,Y)$, value of 1.55 and a limiting reference axial power shape. The Mark-BW MAP limits may be applied to OFA fuel, provided an appropriate adjustment factor is applied to provide equivalence to a 1.49 design $F\Delta H(X,Y)$, for the OFA. This is reflected in the MAP limits specified in the COLR. The relaxation of $F\Delta H(X,Y)$, as a function of THERMAL POWER allows changes in the radial power for all permissible control bank insertion limits. This relaxation is implemented by the application of the following factors:

$$k = [1 + (1/RRH)\ (1 - P)]$$

where k = power factor multiplier applied to the MAP limits

P = THERMAL POWER / RATED THERMAL POWER

RRH is given in the COLR

$FQ^M(X,Y,Z)$ and $F\Delta HR^M(X,Y)$ are measured periodically, and comparisons to the allowable limit are made to provide reasonable assurance that the limiting criteria will not be exceeded for operation within the Technical Specification limits of Sections 2.2 (Limiting Safety Systems Settings), 3.1.3 (Movable Control Assemblies), 3.2.1 (Axial Flux Difference), and 3.2.4 (Quadrant Power Tilt Ratio). A peaking margin calculation is performed to provide a basis for decreasing the width of the AFD and $f(\Delta I)$ limits and for reducing THERMAL POWER.

HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Unit 1) (Continued)

When an $FQ^M(X,Y,Z)$ measurement is obtained in accordance with the surveillance requirements of Specification 4.2.2, no uncertainties are applied to the measured peak; the required uncertainties are included in the peaking limit.

When $FQ^M(X,Y,Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2, the measured peak is increased by the radial-local peaking factor to convert it to a local peak. Allowances of 5% for measurement uncertainty and 3% for manufacturing tolerances are then applied to the measured peak.

When an $F\Delta HR^M(X,Y)$ measurement is obtained, regardless of the reason, no uncertainties are applied to the measured peak; the required uncertainties are included in the peaking limit.

3/4.2.4 QUADRANT POWER TILT RATIO (Unit 1)

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A peaking increase that reflects a QUADRANT POWER TILT RATIO of 1.02 is included in the generation of the AFD limits.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q(X,Y,Z)$ is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 2%.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations.

3/4.2.5 DNB PARAMETERS-(UNIT 1)

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a

3/4.2.5 DNB PARAMETERS-(UNIT 1) (Continued)

design limit DNBR throughout each analyzed transient. As noted on Figure 3.2-1, Reactor Coolant System flow rate and THERMAL POWER may be "traded off" against one another (i.e., a low measured Reactor Coolant System flow rate is acceptable if the THERMAL POWER is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relationship defined on Figure 3.2-1 remains valid as long as the limits placed on the nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, in Specification 3.2.3 are maintained. The indicated T_{avg} value and the indicated pressurizer pressure value correspond to analytical limits of 594.8°F and 2205.3 psig respectively, with allowance for measurement uncertainty. When Reactor Coolant System flow rate is measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-1 since a measurement error of 2.1% for Reactor Coolant System total flow rate has been allowed for in determination of the design DNBR value.

The measurement error for Reactor Coolant System total flow rate is based upon performing a precision heat balance and using the result to calibrate the Reactor Coolant System flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in Figure 3.2-1. Any fouling which might bias the Reactor Coolant System flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the Reactor Coolant System flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. Indication instrumentation measurement uncertainties are accounted for in the limits provided in Table 3.2-1.

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.

3/4.2.1 AXIAL FLUX DIFFERENCE (Unit 2)

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q^{RTP} limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

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.

At power levels below APL^{ND}, the limits on AFD are defined in the COLR, i.e., that defined by the RAOC operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g., load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.

At power levels greater than APL^{ND}, two modes of operation are permissible; 1) RAOC, the AFD limits of which are defined in the COLR, and 2) Base Load operation, which is defined as the maintenance of the AFD within a COLR specified band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(z)$ less than its limiting value. To allow operation at the maximum permissible value, the Base Load operating procedure restricts

the indicated AFD to relatively small target band and power swings (AFD target band as specified in the COLR, $APL^{ND} \leq$ 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.

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 (Unit 2)

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. These limits are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

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;

HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Unit 2) (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 the figure specified in the CORE OPERATING LIMITS REPORT (COLR), 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 the figure specified in the COLR, accounts for $F^N_{\Delta H}$ less than or equal to the $F^{RTP}_{\Delta H}$ limit specified in the COLR. 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 compared to the design DNBR limits, there is sufficient 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 and the $W(Z)_{BL}$ function for Base Load Operation are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Unit 2) (Continued)

When Reactor Coolant System flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of the figure specified in the COLR. Measurement errors of 2.1% for Reactor Coolant System total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

The measurement error for Reactor Coolant System total flow rate is based upon performing a precision heat balance and using the result to calibrate the Reactor Coolant System flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in the figure specified in the COLR. Any fouling which might bias the Reactor Coolant System flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the Reactor Coolant System flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated Reactor Coolant System flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation specified on the figure specified in the COLR.

3/4.2.4 QUADRANT POWER TILT RATIO (Unit 2)

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore

QUADRANT POWER TILT RATIO (Unit 2) (Continued)

flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. The normal locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. Alternate locations are available if any of the normal locations are unavailable.

3/4.2.5 DNB PARAMETERS (Unit 2)

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a design limit DNBR throughout each analyzed transient. The indicated T_{avg} value and the indicated pressurizer pressure value correspond to analytical limits of 594.8°F and 2205.3 psig respectively, with allowance for measurement uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. Indication instrumentation measurement uncertainties are accounted for in the limits provided in Table 3.2-1.

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 NRC in accordance with 10 CFR 50.4, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference Limits, target band*, and APL^{ND*} for Specification 3/4.2.1,
5. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), W(Z)**, APL^{ND**} and $W(Z)_{BL}$ for Specification 3/4.2.2, and
6. Nuclear Enthalpy Rise Hot Channel Factor, $F\Delta HR^{L***}$ or, $F_{\Delta H}^{RTP****}$, and Power Factor Multiplier, $MF\Delta H^{****}$, limits for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux

*Reference 5 is not applicable to target band and APL^{ND}.

**References 5 and 6 are not applicable to W(Z), and APL^{ND}, and $W(Z)_{BL}$.

***Reference 1 is not applicable to $F\Delta HR^{L}$.

****Reference 5 is not applicable to $F_{\Delta H}^{RTP}$ and $MF_{\Delta H}$.

CORE OPERATING LIMITS REPORT (Continued)

Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology.)

3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987, (W Proprietary).

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

4. BAW-10152-A, "NOODLE - A Multi-Dimensional Two-Group Reactor Simulator," June 1985.

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

5. BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse-Designed PWR's," June 1989.

   (Methodology for Specifications 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. BAW-10168P, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," September, 1989.
   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC in accordance with 10 CFR 50.4.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 80 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated January 9, 1991 (Ref. 1), Duke Power Company, et al. (Duke or the licensee) requested amendments to the Technical Specifications (TSs) appended to Facility Operating License Nos. NPF-35 and NPF-52 for the Catawba Nuclear Station, Units 1 and 2 (CNS1 and CNS2). The TS changes are primarily for operation of CNS1 during Cycle 6 with reload fuel and safety and operational analysis provided by the Babcock and Wilcox Fuel Company (BWFC). There are also several administrative changes to the CNS1 TSs, and these are also reflected in administrative changes to the CNS2 TSs. Included with Reference 1 were (1) the proposed TS changes along with related Bases changes and a sample Core Operating Limits Report (COLR), (2) a discussion of the technical justification for the changes, and (3) a CNS1 Cycle 6 reload report prepared by BWFC (Ref. 2).

The reload for CNS1 Cycle 6 is one of the first applications of BWFC providing the reload fuel and the operational and safety analysis for a Westinghouse designed reactor. The changes to the CNS1 TSs are largely a result of changing the BWFC operational methodology and analysis. The characteristics of the Cycle 6 core are not significantly different from those which would have resulted from a reload with Westinghouse standard fuel. BWFC operational methodology, while generally similar to Westinghouse methodology, differs in many details. This requires extensive changes to the TSs for the F_Q and $F_{delta\ H}$ power distribution peaking factors as well as some changes to other TSs related to power distribution and departure from nucleate boiling (DNB).

BWFC operational methodologies (and corresponding TSs) for Westinghouse reactors are similar to BWFC methods for B&W reactors, which have been reviewed and approved by the NRC. They differ primarily in surveillance related areas because Westinghouse reactors do not have the fixed incore neutron detectors of B&W reactors to provide continuous detailed power distribution information. BWFC reload design and safety analysis methods for Westinghouse reactors are also similar to those used and approved for B&W reactors.

In addition to the new fuel for Cycle 6, BWFC has provided a reload analysis and a corresponding Reload Report (Ref. 2) describing the fuel, nuclear and thermal hydraulic design and characteristics, the transient and accident analysis justifying the operations of CNS1 for Cycle 6, and TS changes

necessary to accompany the changes to the operational methodology and Cycle 6 specific analyses. The changes to the operating methodology and TSs include changes to the operating limit parameters assigned to the COLR, and a sample COLR indicating the nature of those changes is provided in the Reload Report.

The methodologies used by BWFC for reload design, safety and operational analyses of Westinghouse designed reactors have been documented in topical reports and reviewed and approved by the NRC. This includes both generic reports and reports specific to the Duke Catawba and McGuire reactors. These reports are listed as references in the Reload Report (Ref. 2). Several of the primary generic reports are listed as References 3 through 7 of this review and the primary Duke specific reports in References 8, 9, and 10.

2.0 EVALUATION

The Cycle 6 reload for CNS1 will be the first use of the BWFC Mark-BW fuel assembly in the Duke reactors. This fuel is similar in most respects to the Westinghouse standard fuel assembly design. There will be 72 of these assemblies with an enrichment of 3.55 percent, along with 121 remaining Westinghouse optimized fuel design (OFA) assemblies from the previous cycle. The mixed (transition) core aspects of the loading are similar to previously reviewed mixtures of Westinghouse standard and OFA fuel assemblies, and they have been considered in the review of the Duke specific topical report (e.g., Reference 9). Limits of the enthalpy rise peaking factor for the two fuel types (OFA limits are 96 percent of the Mark-BW limits) have been provided and no additional penalty is needed for CNS1 Cycle 6 (Refs. 9 and 12). The Mark-BW fuel design has been reviewed and approved by the NRC (e.g., References 3 and 8). The open items in the conclusions of Reference 8 were answered in Reference 11. The answers are acceptable and the NRC review is complete and the fuel design approved, as indicated in the staff review in Reference 11.

The nuclear design and parameters for Cycle 6 have been calculated with standard BWFC methodology, based on the NRC approved NOODLE code (Ref. 4). The core design includes a low-leakage fuel pattern, and the cycle length (350 effective full power days) is slightly longer than CNS1 previous cycles. For this design the physics parameters calculated for the cycle, including control rod worths, reactivity coefficients and shutdown margins, fall within expected ranges and are acceptable.

The thermal-hydraulic design analysis used the generically applicable Statistical Core Design and the BWCMV Critical Heat Flux correlation discussed (and approved by the NRC) in References 5 and 6. The Cycle 6 safety limits are based on enthalpy rise peaking factors of 1.55 for the Mark-BW fuel and 1.49 for OFA fuel. The statistical design limit was determined to be 1.345 for the CNS1 core and the thermal design limit was 1.50, providing a thermal margin of 10.3 percent. The analyses used approved models and reasonable input values, and the thermal-hydraulic design evaluation for Cycle 6 is acceptable.

The Duke specific reload transient and accident safety analyses and evaluations were presented in References 9 and 10, with Reference 9 examining all events except loss of coolant accident (LOCA), which is considered in Reference 10. The scope of the events considered is consistent with that addressed in the Final Safety Analysis Report (FSAR) for Catawba. The evaluations considered the effects of mixed (transition) cores using Westinghouse and Mark-BW fuel. The reports, evaluations and analyses have been reviewed and approved by the NRC staff.

The conclusions of the review of Reference 9 listed several conditions to be considered in using the results of the analyses for cycle specific cases. These conditions were responded to in Reference 12 and with recent information on a benchmark calculation of the steamline break without offsite power. A review of these responses (along with the information in the Reload Report) has concluded the indicated conditions have been satisfied, and the analyses of Reference 9 are acceptable for CNS1 Cycle 6.

The conclusions of Reference 10 accepting the LOCA analyses also contain several conditions. However, most of these are indicated as being satisfied for Catawba, and the others were satisfied by the information provided in the Reload Report. Thus, the analysis provided in Reference 10 is acceptable for CNS1 Cycle 6.

The Reload Report presents values of key parameters used in the analyses of References 9 and 10 and compares them to values determined for CNS1 Cycle 6. All values are within limits, and the analyses are therefore acceptable for Cycle 6. Cycle specific statepoint analyses and dose calculations were performed for the events classified as accidents to confirm that the analyses were applicable and to provide cycle specific dose levels. The dose values met all required criteria and are acceptable.

The new core power distribution related core operating limit methodology developed by BWFC for Westinghouse designed reactors using BWFC fuel, reload analysis and cycle operation control, and adopted by Duke for CNS1 for Cycle 6, is the primary cause for most of the significant changes to the CNS1 TSs for this cycle. The methodology was presented in the BWFC topical report BAW-10163 (Ref. 7) and has been previously reviewed and approved by the NRC. The methodology was developed for reactors such as Catawba with BWFC fuel and analysis and is acceptable for use in CNS1 Cycle 6. Reference 7 discusses the methodology, the relevant terminology (changed in a number of respects from previous Catawba-Westinghouse terminology) and the resulting changes to the power distribution TSs. The report provides example TSs for the revised methodology. The primary TS changes resulting from the methodology are to TSs 3/4.2.2 and 3/4.2.3 on F_Q and $F_{delta\ H}$ limits and surveillance. There are less extensive changes to other related TSs. The CNS1 use of the methodology and the changes to the TSs generally follows the description in Reference 7, but there are some changes in terminology for some parameters, some rearrangement of the TS text to what Duke considers to be a more logical pattern, and a few variations in action and surveillance times, usually more conservative or to maintain previously approved times from the Westinghouse TSs. The surveillance methodology for F_Q and $F_{delta\ H}$ departs from the Reference 7 description by bypassing the first tier surveillance (comparison

of measured to predicted design peaking) and proceeding directly with the second tier peaking margin determination. The two tier method would involve performance of the first tier calculation and if its result was not within expected deviations then the second tier calculation of peaking margin would be performed. Bypassing of the first tier calculation is a conservative change since it eliminates a step involving uncertainties in these surveillances. This is an acceptable deviation, as are the other deviations indicated above.

The methodology adopted for CNS1 Cycle 6 for power distribution operational limits and surveillance is based directly on an NRC approved BWFC methodology, applicable to the CNS1 Cycle 6 reactor, and the proposed TSs flow directly from an approved example set of TSs. Deviations have been explained, and the current review has concluded that the methodology change and the expression of that change in the revised TSs is acceptable.

2.1 TECHNICAL SPECIFICATION CHANGES

In addition to the changes in the TSs resulting from the operational methodology, there are changes resulting from necessary changes to the Core Operating Limits Report (COLR) as well as unrelated minor technical and administrative changes. CNS1 has an approved COLR but the change in methodology requires modifications to the TS transferring parameters to the COLR as well as extensive changes to the COLR itself to describe and list the relevant parameters and associated data. Additions to the list of approved topical reports providing the bases of the methodology are also required. Because CNS1 and CNS2 currently have common TSs and only CNS1 is receiving modified TSs at this time, there are administrative TS changes noting that the CNS2 TSs are unchanged, requiring in some cases separate pages for a given TS for the two reactors.

The following changes are proposed by Duke for the CNS1 and CNS2 TSs. Duke has presented, in Reference 1, an extensive, detailed discussion of, and basis for the changes. The review has concluded that this discussion is complete, correct and acceptable. The changes related to the revised operational methodology will not be discussed in detail here since they have been, for the most part, approved (as an example) in the review of Reference 7.

(1) Index, pages III, IV, V, VIII, XIII; administrative changes adding titles and page numbers for new material. This is acceptable.

(2) TS 2.1 Safety Limits; a new Figure 2.1-1 is added for CNS1 for the reactor core safety limits because of the new BWFC safety analysis methodology. The current figure is retained for CNS2. This is acceptable.

(3) Table 2.2-1 Trip Setpoints; there are small changes to the Loop Minimum Flow, Neutron Flux High and Coolant Flow Low Setpoint (both conservative) and various TA, Z and Sensor Error values. These are based on values used in the safety analyses and a reexamination of plant-specific uncertainty values. They apply in general to both units. Values of constants in the Unit 1 OP delta T and OT delta T trip functions were

revised so that f_2 (delta I) could be set to zero (as indicated and approved in Reference 7). (Unit 2 is already set to zero.) The changes are reasonable and acceptable. Reference to RTD bypass was removed to reflect the removal of that system. Removal of the system was evaluated in amendment 40 to license NPF-35 and amendment 33 to license NPF-52, issued on February 17, 1988. This is an acceptable administrative change.

(4) TS 3/4.2.1, Axial Flux Difference; references to Westinghouse features such as RAOC and baseload operation are removed. This is acceptable. These changes are applicable to Catawba Unit 1 only.

(5) TS 3/4.2.2, Heat Flux Hot Channel Factor; and

(6) TS 3/4.2.3, Nuclear Enthalpy Rise Hot Channel Factor; the Westinghouse methodology is removed from both of these TSs and the BWFC methodology inserted. These changes have been discussed previously and are acceptable. The changes apply only to Unit 1.

(7) TS 3/4.2.4, Quadrant Power Tilt; the statement in the LCO about applicability is moved to the Applicability section. The power penalty for excess tilt is changed to reflect a penalty on thermal power only above a tilt ratio of 1.02. This is acceptable because a 1.02 tilt penalty has been included in the design safety analyses. A footnote is added to the Applicability statement indicating it applies after detector calibration after refueling, and a statement that TS 3.0.4 is not applicable is also added. These changes, except for the tilt penalty, are administrative. These changes to TS 3/4.2.4 are acceptable. The changes apply to Unit 1 only.

(8) TS 3/4.2.5, DNB Parameters; the reactor coolant flow rate limit is moved from TS 3.2.3 to this TS, and is incorporated in a new Figure 3.2-1, where it is combined with the power level to provide permitted, restricted or prohibited operating regions. This figure defines trade-offs in power and flow and has been verified by a number of thermal evaluations. It provides comparable margins to those provided in the previous Westinghouse design TS 3.2.3, which is now revised. It is acceptable. The changes apply to Unit 1 only.

(9) The Bases for the TSs which have been changed for Unit 1 have been revised to reflect the new methodology. These revisions present the changes and reasons for the changes in a satisfactory manner and are acceptable. Current Bases are retained for Unit 2.

(10) TS 3/4.5.1.1, Accumulators, 3/4.5.1.2, Upper Head Injection; these have been changed or deleted to reflect the removal of the UHI system. This is applicable to both units. This change is acceptable. The administrative change of the word "pressurizer" to "Reactor Coolant System" is also acceptable.

(11) TS 4.5.2.h, ECCS Subsystems; flows for the centrifugal charging pump lines and for the safety injection pump lines, each with a single pump running are changed slightly (in appropriate directions) to match values

used in the LOCA analyses. The present configuration meets the revised limits. The change is acceptable for both units.

(12) TS 6.9.1.9, COLR; footnotes have been added to indicate that some of the references are only partially applicable, and new references have been added for the new BWFC methodology. These additions are References 4, 7 and BAW-10168P. These changes are acceptable.

2.2 SUMMARY

The NRC staff has reviewed the reports and other reference material submitted for the justification of the operation of Cycle 6 of CNS1, the use of the BWFC methodology for operation of that cycle and the associated TS revisions, and the safety analysis methodologies and results provided by BWFC. Based on this review, the staff has concluded that appropriate material was submitted and that the operations, fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. The TS changes submitted for this reload for CNS1 and for the administrative changes to Units 1 and 2 reflect the necessary modifications for operation of Unit 1 Cycle 6.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change requirements with respect 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 NRC 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 radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (56 FR 20031). 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.

5.0 CONCLUSION

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

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Date: May 31, 1991

5.0 REFERENCES

1. Letter, and attachments, from H. Tucker, Duke Power Company, to USNRC, dated January 9, 1991, "Technical Specification Amendment, Unit 1 Cycle 6 Reload."

2. BAW-2119, "Reload Report, Catawba Unit 1, Cycle 6," October 1990.

3. BAW-10162P-A, "TACO3, Fuel Pin Thermal Analysis Code," Babcock & Wilcox, Lunchburg, Virginia, November 1989.

4. BAW-10152A, "NOODLE -- A Multi-Dimensional Two-Group Reactor Simulator," Babcock & Wilcox, Lynchburg, Virginia, June 1985.

5. BAW-10170P-A, "Statistical Core Design for Mixing Vane Cores," Babcock & Wilcox, Lynchburg, Virginia, December 1988.

6. BAW-10159P-A, "BMCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies," Babcock & Wilcox, Lynchburg, Virginia, July 1990.

7. BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse-Designed PWRs," Babcock & Wilcox, Lynchburg, Virginia, June 1989.

8. BAW-10172P, "Mark-BW Mechanical Design Report," Babcock & Wilcox, Lynchburg, Virginia, July 1988.

9. BAW-10173P, Revision 2, "Mark-BW Reload Safety Analysis for Catawba and McGuire," Babcock & Wilcox, November 1990.

10. BAW-10174, Revision 1, "Mark-BW Reload LOCA Analysis for the Catawba and McGuire Units," Babcock & Wilcox, November 1990.

11. DPC-NE-2001P-A, "Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel," Revision 1, January 1990, Approved October 1990.

12. Letter and attachment from M. Tuckman, Duke Power Company, to USNRC, dated March 14, 1991, "Response to Conditions Relative to the Use of Topical Report BAW-10173."

DATED: May 31, 1991

AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NPF-35 - Catawba Nuclear Station, Unit 1
AMENDMENT NO. 80 TO FACILITY OPERATING LICENSE NPF-52 - Catawba Nuclear Station, Unit 2

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