October 18, 1989

Docket No. 50-400

DISTRIBUTION
See attached list

Mr. Lynn W. Eury
Executive Vice President
Power Supply
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Eury:

SUBJECT: ISSUANCE OF AMENDMENT NO. 15 TO FACILITY OPERATING LICENSE NO. NPF-63 - SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1, REGARDING CYCLE 3 RELOAD WITH VANTAGE 5 FUEL (TAC NO. 72947)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 15 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1. This amendment consists of changes to the Technical Specifications in response to your request dated April 17, 1989, as supplemented June 29, 1989, July 13, 1989 and October 2, 1989.

The amendment changes the Technical Specifications to allow refueling and operation with Vantage 5 fuel design.

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

Sincerely,

Original Signed By:

Richard A. Becker, Project Manager
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 15 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

CP-1

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:D:PD21:DRPR	:	:	:
NAME	:PAnderson:	: RBecker	: EAdensam	:	:	:
DATE	: 10/11/89	: 10/11/89	: 10/11/89	:	:	:

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AMENDMENT NO. 15 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

Docket File
NRC PDR
Local PDR
PDII-1 Reading
S. Varga (14E4)
G. Lainas
E. Adensam
P. Anderson
R. Becker
OGC
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
B. Grimes (9A2)
T. Meeks (4) (P1-137)
W. Jones (P-130A)
J. Calvo (11D3)
J. Guo (8D1)
S. Sun (8E23)
ACRS (10)
GPA/PA
ARM/LFMB

cc: Licensee/Applicant Service List

Mr. L. W. Eury
Carolina Power & Light Company

Shearon Harris

cc:

Mr. R. E. Jones, General Counsel
Carolina Power & Light Company
P. O. Box 1551
Raleigh, North Carolina 27602

Resident Inspector/Harris NPS
c/o U. S. Nuclear Regulatory Commission
Route 1, Box 315B
New Hill, North Carolina 27562

Mr. R. B. Richey, Manager
Harris Nuclear Project
Harris Nuclear Plant
P. O. Box 165
New Hill, North Carolina 27562

Mr. H. A. Cole
Special Deputy Attorney General
State of North Carolina
P. O. Box 629
Raleigh, North Carolina 27602

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

Mr. C. S. Hinnant
Plant General Manager
Harris Nuclear Plant
P. O. Box 165
New Hill, North Carolina 27562

Mr. Dayne H. Brown, Chief
Radiation Protection Section
Division of Facility Services
N. C. Department of Human Resources
701 Barbour Drive
Raleigh, North Carolina 27603-2008

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 15
License No. NPF-63

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Carolina Power & Light Company, (the licensee), dated April 17, 1989, as supplemented June 29, 1989, July 13, 1989 and October 2, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-63 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 15, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 18, 1989

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	: OGC	:D:PD21:DRPR	:	:	:
NAME	:PAnderson	: RBecker	: E. Chan	: EAdensam	:	:	:
DATE	: 10/11/89	: 10/11/89	: 10/13/89	: 10/17/89	:	:	:

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ATTACHMENT TO LICENSE AMENDMENT NO. 15

FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
i	i
iv	iv
v	v
xix	xix
1-2	1-2
-	1-2a
2-2	2-2
2-4	2-4
2-5	2-5
2-7	2-7
2-8	2-8
2-9	2-9
2-10	2-10
B 2-1	B 2-1
-	B 2-1a
B 2-4	B 2-4
B 2-5	B 2-5
B 2-6	B 2-6

Remove Pages	Insert Pages
3/4 1-14	3/4 1-14
3/4 1-19	3/4 1-19
3/4 1-20	3/4 1-20
3/4 1-21	3/4 1-21
3/4 1-22	3/4 1-22
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-4	3/4 2-4
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3/4 2-7c	3/4 2-7c
3/4 2-7d	3/4 2-7d
3/4 2-8	3/4 2-8
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3/4 2-14	3/4 2-14
3/4 3-7	3/4 3-7
3/4 3-34	3/4 3-34
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-2a	B 3/4 2-2a
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5

Remove Pages	Insert Pages
B 3/4 2-6	B 3/4 2-6
B 3/4 4-1	B 3/4 4-1
6-24	6-24
6-24a	6-24a

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1.0 DEFINITIONS

INDEX

3.0/4.0 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

ADMINISTRATIVE CONTROLS

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9.a The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Plant operation within these core operating limits is addressed within the individual specifications.

DIGITAL CHANNEL OPERATIONAL TEST

1.10 A DIGITAL CHANNEL OPERATIONAL TEST shall consist of exercising the digital computer hardware using data base manipulation to verify OPERABILITY of alarm and/or trip functions.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

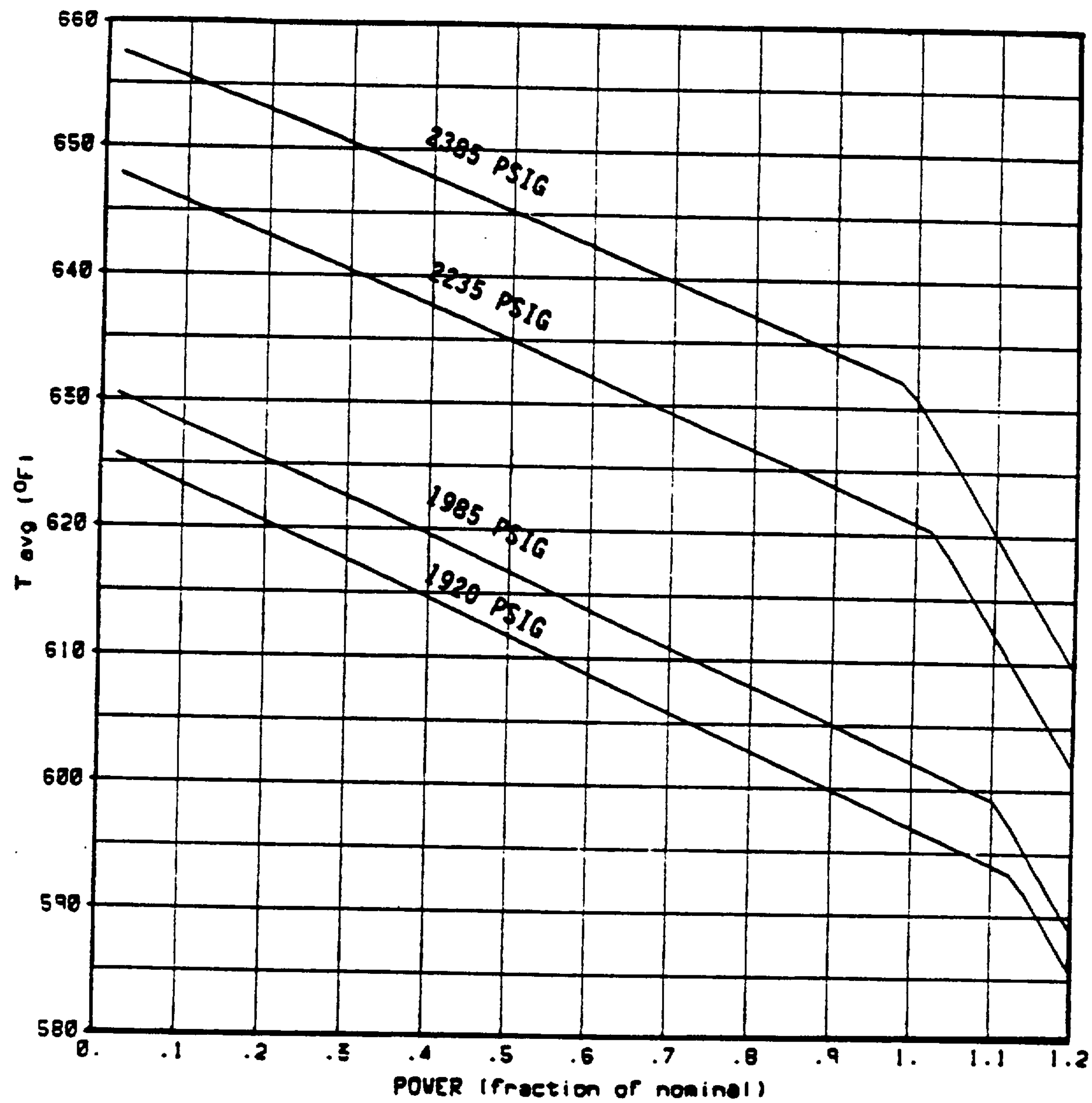

FIGURE 2.1-1
REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION

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	4.56	0	$\leq$109% of RTP**	$\leq$111.1% of RTP**
b. Low Setpoint	8.3	4.56	0	$\leq$25% of RTP**	$\leq$27.1% of RTP**
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	$\leq$5% of RTP** with a time constant $\geq$2 seconds	$\leq$6.3% of RTP** with a time constant $\geq$2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	$\leq$5% of RTP** with a time constant $\geq$2 seconds	$\leq$6.3% of RTP** with a time constant $\geq$2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	$\leq$25% of RTP**	$\leq$30.9% of RTP**
6. Source Range, Neutron Flux	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT	8.7	6.02	Note 5	See Note 1	See Note 2
8. Overpower ΔT	4.7	1.50	1.9	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	2.21	1.5	$\geq$1960 psig	$\geq$1946 psig
10. Pressurizer Pressure-High	7.5	5.01	0.5	$\leq$2385 psig	$\leq$2399 psig
11. Pressurizer Water Level-High	8.0	2.18	1.5	$\leq$92% of instrument span	$\leq$93.8% of instrument span

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	2.9	1.98	0.6	≥90.5% of loop full indicated flow	≥89.5% of loop full indicated flow
13. Steam Generator Water Level Low-Low	19.2	18.18	1.5	≥38.5% of narrow range instrument span	≥38.0% of narrow range instrument span
14. Steam Generator Water Level-Low Coincident With	19.2	6.68	1.5	≥38.5% of narrow range instrument span	≥36.8% of narrow range instrument span
Steam/Feedwater Flow Mismatch	20.0	3.41	Note 6	≤40% of full steam flow at RTP**	≤43.1% of full steam flow at RTP**
15. Undervoltage - Reactor Coolant Pumps	14.0	1.3	0.0	≥5148 volts	≥4920 volts
16. Underfrequency - Reactor Coolant Pumps	5.0	3.0	0.0	≥57.5 Hz	≥57.3 Hz
17. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥1000 psig	≥950 psig
b. Turbine Throttle Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A	N.A.

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where: ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ s, $\tau_2 = 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.17;
K_2	=	0.0224/°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}, $\tau_4 = 20$ 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, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: (Continued)

T' $\leq$ 588.8°F (Nominal T_{avg} at RATED THERMAL POWER);

K_3 = 0.001072/psig;

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

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

(1) For $q_t - q_b$ between -21.6% and +6.0%, f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -21.6%, the ΔT Trip Setpoint shall be automatically reduced by 2.36% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 6.0%, the ΔT Trip Setpoint shall be automatically reduced by 1.57% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.1% ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{(1 + \tau_3 S)}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{(1 + \tau_7 S)}\right) \left(\frac{1}{(1 + \tau_6 S)}\right) T - K_6 \left[T \left(\frac{1}{(1 + \tau_6 S)}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1 + \tau_3 s}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_o = As defined in Note 1,

K_4 = 1.079,

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: (Continued)

K_6 = 0.002/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$,

T = As defined in Note 1,

T'' = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.8°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 2.3% ΔT span.

NOTE 5: The sensor error for temperature is 1.9 and 1.1 for pressure.

NOTE 6: The sensor error for steam flow is 0.9, for feed flow is 1.5, and for steam pressure is 0.75.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

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

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation 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) defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent 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 or WRB-2 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for the WRB-1 or WRB-2 correlation).

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 percent probability with 95 percent confidence level 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. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

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.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE (Continued)

These curves are based on an enthalpy hot channel factor, $F_{\Delta H}$, of 1.62 for LOPAR fuel and 1.65 for VANTAGE 5 fuel and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in calculated $F_{\Delta H}$ at reduced power based on the expression:

$$F_{\Delta H} = 1.62 \, [1 + 0.3 \, (1-P)] \text{ for LOPAR fuel, and}$$

$$F_{\Delta H} = 1.65 \, [1 + 0.35 \, (1-P)] \text{ for VANTAGE 5 fuel}$$

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.

Power Range, Neutron Flux (Continued)

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than the design DNBR value.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors , (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Low Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90.5% of nominal full loop flow. Above P-8

Reactor Coolant Flow (Continued)

(a power level of approximately 49% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90.5% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to 1.627×10^6 lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 38.5%, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Buses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.3 second.

On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one rod misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With more than one rod inoperable, due to a rod control urgent failure alarm or obvious electrical problem in the rod control system existing for greater than 36 hours, be in HOT STANDBY within the following 6 hours.

d. With one rod trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits on control banks specified in the Core Operating Limits Report. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with two reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of shutdown and control rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn as specified in the Core Operating Limits Report.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.
**With K_{eff} greater than or equal to 1.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the Core Operating Limits Report.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

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

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

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the Core Operating Limits Report, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the rod insertion limit monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

FIGURE 3.1-2 DELETED.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

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

a. the acceptable operational space as specified in the Core Operating Limits Report for Relaxed Axial Offset Control (RAOC) operation, or

b. within a band about the target AFD during Base Load operation as specified in the Core Operating Limits Report.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

a. For RAOC operation with the indicated AFD outside of the limits specified in the Core Operating Limits Report, either:

   1. Restore the indicated AFD to within the limits specified in the Core Operating Limits Report within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. For Base Load operation above APL^{ND}** with the indicated AXIAL FLUX DIFFERENCE outside of the applicable target band about the target AFD, either:

   1. Restore the indicated AFD to within the target band limits within 15 minutes, or

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

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the Core Operating Limits Report for RAOC operation.

*See Special Test Exception 3.10.2

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

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 at least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

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

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

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

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

FIGURE 3.2-1 DELETED

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

LIMITING CONDITION FOR OPERATION

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

$$F_Q(Z) \leq \frac{2.45}{P} [K(Z)] \text{ FOR } P > 0.5$$

$$F_Q(Z) \leq (4.90) [K(Z)] \text{ FOR } P \leq 0.5$$

Where:

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

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

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit.

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

c. Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{2.45}{P \times W(Z)} \times K(Z) \text{ for } P > 0.5$$

$$F_Q^M(Z) \leq \frac{2.45}{W(Z) \times 0.5} \times K(Z) \text{ for } P \leq 0.5$$

where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.45 is the F_Q limit, K(Z) is given in Figure 3.2-2, P is the fraction of RATED THERMAL POWER, and W(Z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Core Operating Limits Report as per Specification 6.9.1.6.

d. Measuring $F_Q^M(Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined,* or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

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

SURVEILLANCE REQUIREMENTS (Continued)

e. With measurements indicating

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

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

1) $F_Q^M(Z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c, or

2) $F_Q^M(Z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

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

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:

$$\left\{ \begin{array}{ll} \left(\text{maximum} \left[\dfrac{F_Q^M(Z) \times W(Z)}{\dfrac{2.45}{P} \times K(Z)} \right] \right) - 1 & \times\ 100 \text{ for } P \geq 0.5 \\ \left(\text{maximum} \left[\dfrac{F_Q^M(Z) \times W(Z)}{\dfrac{2.45}{0.5} \times K(Z)} \right] \right) - 1 & \times\ 100 \text{ for } P < 0.5 \end{array} \right.$$

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 specified in the Core Operating Limits Report by 1% AFD for each percent $F_Q(Z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within the limits specified in the Core Operating Limits Report) during this time period. Base Load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and FQ surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

$$APL^{BL} = \text{minimum} \left[\frac{2.45 \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}} \right] \times 100\%$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is 2.45. K(Z) is given in Figure 3.2-2. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during Base Load operation. The function is given in the Core Operating Limits Report as per Specification 6.9.1.6.

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

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

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND}.

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

SURVEILLANCE REQUIREMENTS (Continued)

c. Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{2.45 \times K(Z)}{P \times W(Z)_{BL}} \quad \text{for } P > APL^{ND}$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. The F_Q limit is 2.45.

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

d. Measuring $F_Q^M(Z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering Base Load operation after satisfying Section 4.2.2.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and

2. At least once per 31 effective full power days.

e. With measurements indicating

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

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

1. $F_Q^M(Z)$ shall be increased by 2 percent over that specified in 4.2.2.4.c, or

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

$$[(\text{max. of } [\frac{F_Q^M(Z) \times W(Z)_{BL}}{\frac{2.45}{P} \times K(Z)}]) - 1] \times 100 \text{ for } P \geq APL^{ND}$$

g. The limits specified in 4.2.2.4.c, 4.2.2.4.e, and 4.2.2.4.f above are not applicable in the following core plane regions:

1. Lower core region 0 to 15 percent, inclusive.

2. Upper core region 85 to 100 percent, inclusive.

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

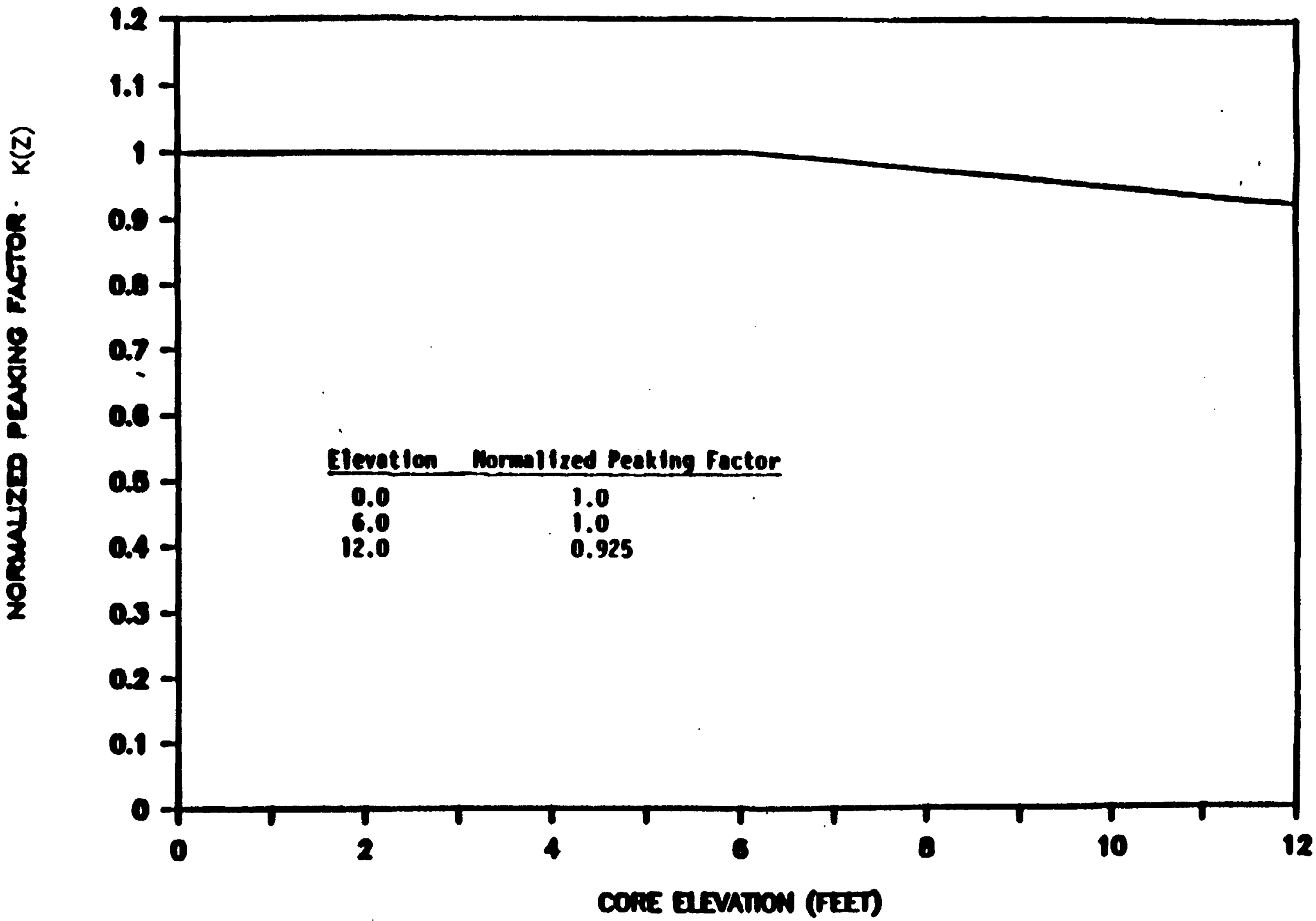

FIGURE 3.2-2
K(Z) - LOCAL AXIAL PENALTY FUNCTION FOR F_Q (Z)

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

LIMITING CONDITION FOR OPERATION

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

a. Measured RCS flow rate $\geq$ 293,540 gpm x (1.0 + C_1), and

b. $F_{\Delta H} \leq 1.62\ [1.0 + 0.3(1.0-P)]$ for LOPAR fuel, and

$F_{\Delta H} \leq 1.65\ [1.0 + 0.35(1.0-P)]$ for VANTAGE 5 fuel.

Where:

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

$F_{\Delta H}$ = Nuclear enthalpy rise hot channel factor obtained by using the movable incore detectors to obtain a power distribution map, with the measured value of the nuclear enthalpy rise hot channel factor ($F^N_{\Delta H}$) increased by an allowance of 4% to account for measurement uncertainty.

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

APPLICABILITY: MODE 1.

ACTION:

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

a. Within 2 hours either:

1. Restore RCS total flow rate and $F_{\Delta H}$ to within the above limits, or

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

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 following limits:

a. Indicated Reactor Coolant System $T_{avg} \leq 594.1°F$ after addition for instrument uncertainty, and

b. Indicated Pressurizer Pressure $\geq$ 2185 psig* after subtraction for instrument uncertainty.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its indicated 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 shown in Specification 3.2.5 shall be verified to be within its limit at least once per 12 hours.

*This limit is not applicable during either a Thermal Power Ramp in excess of ±5% Rated Thermal Power per minute or a Thermal Power step change in excess of ±10% Rated Thermal Power.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers, and verify compliance with the shutdown margin requirements of Specification 3.1.1.2 within 1 hour and at least once per 12 hours thereafter.

b. With no channels OPERABLE, open the Reactor Trip System breakers within 1 hour and suspend all operations involving positive reactivity changes. Verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.2 within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Loss-of-Offsite Power					
a. 6.9 kV Emergency Bus Undervoltage--Primary	N.A.	N.A.	N.A.	$\geq$ 4830 volts with a $\leq$ 1.0 second time delay.	$\geq$ 4692 volts with a time delay $\leq$ 1.5 seconds
b. 6.9 kV Emergency Bus Undervoltage--Secondary	N.A.	N.A.	N.A.	$\geq$ 6420 volts with a $\leq$ 16 second time delay (with Safety Injection).	$\geq$ 6392 volts with a time delay $\leq$ 18 seconds (with Safety Injection).
				$\geq$ 6420 volts with a $\leq$ 54.0 second time delay (with-out Safety Injection).	$\geq$ 6392 volts with a $\leq$ 60 second time delay (with-out Safety Injection).
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure,					
P-11	N.A.	N.A.	N.A.	$\geq$ 2000 psig	$\geq$ 1986 psig
Not P-11	N.A.	N.A.	N.A.	$\leq$ 2000 psig	$\leq$ 2014 psig
b. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	$\geq$ 553°F	$\geq$ 549.3°F

3/4.2 POWER DISTRIBUTION LIMITS

BASES

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

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

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

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

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.45 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

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

AXIAL FLUX DIFFERENCE (Continued)

At power levels below APL^{ND}, the limits on AFD are specified in the Core Operating Limits Report, 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 with fixed AFD limits as a function of reactor power level and 2) Base Load operation which is defined as the maintenance of the AFD within a band about a target value. Both the fixed AFD limits for RAOC operation and the band for Base Load operation are specified for each reload cycle in the CORE OPERATION LIMITS REPORT per Specification 6.9.1.6. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(Z)$ less than its limiting value. To allow operation at the maximum permissible value, the Base Load operating procedure restricts the indicated AFD to a relatively small target band and power swings. For Base Load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24-hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period, load changes and rod motion are restricted to that allowed by the Base Load procedure. After the waiting period, extended Base Load operation is permissible.

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

3/4.2.2 AND 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, RCS FLOW RATE, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flow rate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

HEAT FLUX HOT CHANNEL FACTOR, AND RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

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

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

$F_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The combinations of the RCS flow requirement and the measurement of $F_{\Delta H}$ ensures that the calculated DNBR will not be below the design DNBR value. The relaxation of $F_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

$F^N_{\Delta H}$ is evaluated as being less than or equal to 1.56 for LOPAR fuel and 1.59 for VANTAGE 5 fuel. These values are 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.

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is more than sufficient to offset any rod bow penalty and transition core penalty.

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

The hot channel factor $F^M_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either RAOC or Base Load operation, $W(Z)$ or $W(Z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(Z)$, is met. $W(Z)$ accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(Z)_{BL}$ accounts for the more restrictive operating limits allowed by Base Load operation which result in less severe transient values. The $W(Z)$ function for normal operation is provided in the Core Operating Limits Report per Specification 6.9.1.6.

HEAT FLUX HOT CHANNEL FACTOR, AND RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

A measurement error of 4% for $F^N_{\Delta H}$ has been allowed for in determination of the design DNBR value. When RCS flow rate is measured, no additional allowance is necessary prior to comparison with the limit of Specification 3.2.3. A normal RCS flowrate error of 2.1% will be included in C_1, which will be modified as discussed below.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS 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 non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi, raises the nominal flow measurement allowance, C_1, to 2.2% for no venturi fouling. Any fouling which might bias the RCS 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 RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation that could lead to operation outside the acceptable region of operation.

3/4.2.4 QUADRANT POWER TILT RATIO

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 limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. 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.

QUADRANT POWER TILT RATIO (Continued)

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 preferred sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. If other locations must be used, a special report to NRC should be submitted within 30 days in accordance with 10CFR50.4.

3/4.2.5 DNB PARAMETERS

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 minimum DNBR that is equal to or greater than the design DNBR value throughout each analyzed transient. The indicated T_{avg} value and the indicated pressurizer pressure value are compared to analytical limits of 594.1°F and 2185 psig, respectively, after an allowance for measurement uncertainty is included.

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.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design DNBR value during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 335°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 380,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

6.9.1.6 CORE OPERATING LIMITS REPORT

6.9.1.6.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. The shutdown rod insertion limits of Specification 3.1.3.5.

b. The control rod insertion limits of Specification 3.1.3.6.

c. The axial flux difference of Specification 3.2.1.

d. The surveillance requirements of Specifications 4.2.2.2, 4.2.2.3, and 4.2.2.4.

6.9.1.6.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. WCAP-10216-P-A, Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification, 1983.

b. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, 1985.

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

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

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

6.10 RECORD RETENTION

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

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

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

RECORD RETENTION (Continued)

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

c. All REPORTABLE EVENTS;

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

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

f. Records of radioactive shipments;

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 15 TO FACILITY OPERATING

LICENSE NO. NPF-63

CAROLINA POWER & LIGHT COMPANY, et. al

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By letter dated April 17, 1989 (Reference 1), as supplemented by letters dated June 29, 1989, July 13, 1989 and October 2, 1989, Carolina Power & Light Company (CP&L) submitted a request for Technical Specification (TS) changes to allow refueling and operation of the Shearon Harris Nuclear Power Plant, Unit 1, (Harris) Cycle 3 core with the VANTAGE 5 fuel design. The June 29, 1989, July 13, 1989 and October 2, 1989 letters provided clarifying information that did not alter the action noticed, or change the initial determination of no significant hazards consideration as published in the <u>Federal Register</u> on July 12, 1989 (54 FR 29399).

Currently, Harris is operating with a Westinghouse 17x17 low parasitic (LOPAR) fueled core. Future core loadings will consist of a mixed core of a 67% LOPAR and 33% VANTAGE 5 in the Cycle 3 transition core to eventually an all VANTAGE 5 fueled core. The VANTAGE 5 fuel design has been approved with conditions in the NRC safety evaluation on Westinghouse topical report WCAP-10444-P-A, "Reference Core Report VANTAGE 5 Fuel Assembly." The major design features of the VANTAGE 5 fuel relative to the current LOPAR fuel include the following: (1) integral fuel burnable absorber (IFBA), (2) intermediate flow mixer (IFM) grids, (3) reconstitutable top nozzle (RTN), (4) extended burnup capability, (5) axial blankets, and (6) smaller diameters in fuel rod, guide thimble and instrumentation. The licensee indicated (Reference 2) that the transition core and full VANTAGE 5 core safety analyses were performed at a thermal power level of 2775 MWt with assumptions including a full power F_H of 1.62 for the LOPAR fuel and 1.65 for the VANTAGE 5 fuel, a maximum F_Q of 2.45, 6% steam generator tube plugging for the LOCA analysis, 0% for the non-LOCA analysis, and a positive moderator temperature coefficient of +5 PCM/°F from 0 to 70% power, decreasing linearly to 0 PCM/°F between 70 to 100% power.

The TS changes include: (1) use of Westinghouse WRB-2 DNBR correlation for the VANTAGE 5 fuel, (2) an increased maximum F_H from 1.62 to 1.65, (3) an increased control rod drop time from 2.2 to 2.7 seconds, (4) revised overpower/overtemperature setpoints, (5) use of the core operating limits report to document the cycle specific operating limits, and (6) Cycle 2 administrative corrections.

During the review of the VANTAGE 5 fuel design in WCAP-10444-P-A, the staff identified a few conditions to be resolved for licensees using the VANTAGE 5 fuel design. The staff's review of the licensee's request for the TS changes and the associated supporting analyses (References 2 through 7 and 9) will address those conditions listed in the Safety Evaluation (SE) on WCAP-10444-P-A in the following evaluation.

2.0 EVALUATION

2.1 Statistical Convolution Method

In the SE on WCAP-10444-P-A, the staff stated that the statistical method should not be used in VANTAGE 5 for evaluating the fuel rod shoulder gap. The licensee indicated (Reference 8) that the statistical convolution method was not used for the VANTAGE 5 fuel design and the currently approved method was used for evaluating the fuel rod shoulder gap. Therefore, the staff considers this acceptable.

2.2 Seismic and LOCA Loads

In the SE on WCAP-10444-P-A, the staff stated that for each plant application, it must be demonstrated that the VANTAGES 5 fuel assembly will maintain its coolable geometry under combined seismic and LOCA loads. The licensee performed LOCA and seismic load analyses for a transition core and an all VANTAGE 5 core using an approved method. The results (Reference 8) showed that the fuel assembly in either case has enough margin to sustain the combined seismic and LOCA loads such that the structural integrity and coolable geometry are maintained. Based on the licensee's acceptable results, the staff concludes that the condition of seismic and LOCA loads is satisfied.

2.3 Irradiation Demonstration Program

In the SE on WCAP-10444-P-A, the staff required that an irradiation program be performed to confirm the VANTAGE 5 fuel performance. The licensee indicated that there were numerous demonstration programs involving Westinghouse OFA fuel assemblies containing Zircaloy grids irradiated in 17x17 cores. The satisfactory performance of these demonstration assemblies resulted in OFA with Zircaloy grids reload in many Westinghouse reactors. The OFA fuel assemblies with Zircaloy grids cover the VANTAGE 5 fuel design features. Thus, the staff concludes that VANTAGE 5 assemblies will perform satisfactorily in Harris.

2.4 Improved Thermal Design Procedure (ITDP)

In the SE on WCAP-10444-P-A, the staff stated that the ITDP was approved with restrictions (References 10 and 11) as applied to the VANTAGE 5 fuel design. The licensee indicated (Reference 8) that they complied with the ITDP restrictions for Harris. The staff, therefore, concludes that ITDP was acceptably applied to the Harris VANTAGE 5 cores.

2.5 Positive Moderator Temperature Coefficient

In the SE on WCAP-10444-P-A, the staff stated that if a positive moderator temperature coefficient (MTC) is intended, the same MTC should be used in the plant specific analysis. The licensee indicated that a positive MTC was considered in the analysis. The staff thus concludes that this restriction is satisfactorily met.

2.6 Transient Analysis

In the SE on WCAP-10444-P-A, the staff requires that a plant specific analysis be performed to show that the appropriate safety criteria are not violated with the higher value of F_H and use of the VANTAGE 5 fuel. The licensee reanalyzed the transients affected significantly by the fuel design update and operating condition changes. In References 1 and 3, the staff finds that no acceptance criteria are violated for any of the transients and accidents reanalyzed and the Steam Line Break mass and energy releases previously calculated are not adversely affected by the transition to VANTAGE 5 fuel. The staff, therefore, concludes that the transient and accidents analyses are acceptable.

2.7 LOCA Analysis

In the SE on WCAP-10444-P-A, the staff stated that the plant specific analysis should be performed to show that the requirements of 10 CFR 50.46 are met. The licensee analyzed large break LOCAs and small break LOCAs to support the reload licensing application. In the licensee's large break LOCA analysis (Reference 5) only double ended cold leg guillotine (DECLG) breaks were analyzed for a full core of VANTAGE 5 fuel (instead of LOPAR fuel) since they were identified previously as limiting cases that result in the highest peak cladding temperature (PCT). The DECLG break analysis was performed with a total peaking factor of 2.45, 102% of the core power of 2775 MWt and an assumed loss of offsite power at the beginning of the accident. A sensitivity study of DECLG break sizes on the effect of the PCT was performed by use of discharge coefficients (C_D) of 0.8, 0.6 and 0.4. The results showed that the DECLG break with a discharge coefficient of 0.4 is the worst large break case, resulting in a PCT of 2105.2°F. The analysis was performed with a modified version of the 1981 Westinghouse ECCS evaluation model (Reference 12). This evaluation model used the revised PAD fuel thermal safety model (Reference 13) for the calculation of the initial fuel conditions; the SATAN-VI code (Reference 14) for the transient thermal hydraulic calculation during the blowdown period; the WREFLOOD (Reference 15) and BASH (Reference 16) codes for the thermal hydraulic calculation during refill and reflood transient periods; the LOCBART code (Reference 16) for calculation of the peak cladding temperature and the COCO code (Reference 17) for the calculation of containment pressure transient.

For the mixed core, the VANTAGE 5 fuel differs hydraulically from the LOPAR assembly design it replaces. The difference in the total assembly hydraulic resistance between the two designs is approximately 10 percent higher for VANTAGE 5. The higher flow resistance results in a reduction in flow rate for the VANTAGE 5 fuel assembly during mixed core conditions. The licensee indicated that the maximum PCT increase due to flow rate decrease is 50°F.

As a result of our review, the staff found that the approved analytical models and computer codes were used and results showed that the peak cladding temperature metal-water reaction and clad oxidation are within the acceptance criteria imposed in 10 CFR 50.46 for a LOCA analysis. Therefore, the staff concludes that the large break LOCA analysis is acceptable.

In the licensee's small break analysis, the staff finds that the NRC approved NOTRUMP code (References 18 and 19) was used for the calculation of transient core power and depressurization of the reactor coolant system. Only one core flow channel is modeled in NOTRUMP since the core flow during a small break is relatively slow providing enough time to maintain flow equilibrium between fuel assemblies in mixed cores (i.e., no cross flow). Hydraulic resistance mismatch is not a factor for a small break. Therefore, the licensee referenced the small break LOCA for the complete core of the VANTAGE 5 fuel design as bounding for all transition cycles. The LOCTA code (Reference 20) was used for the PCT calculation. The analysis was done with assumptions of 102% of the core power of 2775 MWt and a total peaking factor of 2.50. Analyses for three break sizes were performed to show that the worst break size is a 3-inch diameter break which results in the highest PCT of 1780°F, well below the acceptance criterion of 2200°F. The staff, therefore, concludes that the small break LOCA analysis is acceptable since the approved methods were used to show the analytical results to be within the acceptance criteria in 10 CFR 50.46.

In the licensee's steam generator tube rupture (SGTR) analysis, two cases were analyzed. Case 1 is a SGTR with an assumption of the failure of an intact SG power operated relief valve (PORV) to open on demand when cooldown of the reactor coolant system is initiated. Case 2 is a SGTR with the PORV of the ruptured SG PORV is assumed to fail open when the isolation of the ruptured SG is performed. The licensee indicated that they used the NRC approved LOFTTR2 code (Reference 8) for the SGTR analysis and used the calculated steam release resulting from SGTR dose calculation for radiological impact assessment to show that the results are within the allowable guidelines specified in the Standard Review Plan (NUREG-800, Section 15.6.3) and 10 CFR 100. The staff finds that the licensee's approach is adequate and concludes that the SGTR analysis is acceptable.

2.8 Reactor Coolant Pump Shaft Seizure

In the SE on WCAP-10444-P-A, the staff stated that the mechanistic approach in determining the fraction of the fuel failures during the reactor coolant pump shaft seizure accident was unacceptable and the fuel failure criteria should be 95/95 DNBR limit. Using the approved method, the licensee reanalyzed this accident for Harris. The results showed that the number of fuel failures was less than 30% of the total fuel rods in the core based on 95/95 DNBR limit. The amount of fuel failure is used in the dose calculation for the radiological consequences assessment. Based on the acceptable fuel failure criterion of 95/95 DNBR limit, the staff concludes that the reactor coolant pump shaft seizure event is satisfactorily addressed for VANTAGE 5 fuel.

2.9 Design Basis Accident and Radiological Consequences Relative to VANTAGE 5 Fuel

In the SE for License Amendment No. 7, the staff examined the impact on the Design Basis Accidents (DBA) of raising fuel enrichment to 4.2 weight percent U-235 and fuel burnup to 60,000 MWD/MT. In the SE for the License Amendment No. 7, the staff concluded that the DBA previously analyzed by the licensee in their Final Safety Analysis Report (FSAR) bounded any potential radiological consequences of DBA that could result with extended burnup and 4.2 weight percent U-235 fuel. The licensee in this proposal has requested the use of up to 5.0 weight percent U-235 fuel. Increasing the enrichment from 4.2 to 5.0

weight percent of U-235 has only a minimal effect on the DBA radiological consequences compared to the effect of increased burnup. Therefore, the FSAR DBA radiological consequences also bound the VANTAGE 5 DBA radiological consequences and the VANTAGE 5 DBA radiological consequences are acceptable.

3.0 TECHNICAL SPECIFICATION CHANGES

To use the Cycle 3 VANTAGE 5 core, the licensee submitted a request for numerous TS changes. The TS changes reflect the impact of the fuel design, analytical methods and assumptions used in the safety analysis to support the reload application. The TS changes are discussed individually in the following paragraphs. The respective changes are also identified by TS page number(s).

3.1 Core Safety Limits (Figure 2.1-1, 3/4.2.3 and Bases 2.1.1, 3/4.2, 3/4.2.5, 3/4.4.1)

The core safety limits are revised to reflect the use of VANTAGE 5, ITDP, WRB-1 (LOPAR fuel) and WRB-2 (VANTAGE 5 fuel) DNBR correlation and increased F_H (1.62 for LOPAR fuel and 1.65 for VANTAGE 5 fuel). Since the core safety limits are calculated based on the approved WCAP-10444-P-A, ITDP method, DNBR correlations and acceptable results of transient and accident analyses, the staff concludes that the revised core safety limits are acceptable.

3.2 Overpower/Overtemperature Setpoints (Figure 2.1-1, Table 2.2-1)

Since overpower and Overtemperature setpoints are revised to maintain consistency with the analyses to support the reload application, the revised overpower and overtemperature setpoints are acceptable.

3.3 Increased RCCA Drop Times (3.1.3.4)

The RCCA drop time is revised to be less than 2.7 seconds from 2.2 seconds due to the use of the VANTAGE 5 fuel design. The licensee has taken into account the effect of the increased RCCA drop time in all safety analyses. The staff concludes this change is acceptable.

3.4 RCS Design Flow Change (3.2.3)

The RCS design flow is increased from 292,800 to 293,540 gpm. The change is consistent with the assumption used for the transient analyses and is acceptable.

3.5 Hot Channel Factor-F_Q (3.2.2, 4.2.2.2, 4.2.2.3, 4.2.2.4, Figure 3.2-2 and Bases 3/4.2)

The maximum hot channel factor-F_Q is increased from 2.32 to 2.45 to allow for greater operational flexibility due to the use of VANTAGE 5 fuel. Since the F_Q of 2.45 was used in the analysis to support the reload application, the staff concludes this change is acceptable.

3.6 DNB Parameters (3/4.2.5)

The proposed limits on DNB related parameters (Tavg and pressurizer pressure) assure that the parameters are maintained within the normal steady state envelope of operation. Since the revisions are consistent with the assumptions used in the transient and accident analyses, the staff concludes the changes are acceptable.

3.7 Cycle 2 Corrections (4.2.1.1, Table 3.3-1)

Two administrative corrections were included with the TS changes requested. They are: (1) deletion of surveillance requirement 4.2.1.1.a.2 which requires monitoring and logging of indicated Axial Flux Difference for a 24-hour period after the automatic monitoring is returned to an operable status, and (2) deletion of reference to LCO 3.1.1.1. These changes are consistent with the changes approved by NRC for Cycle 2 operation and are, therefore, acceptable.

3.8 Trip Setpoint and Uncertainty Changes (Table 3.3-4, Table 3.4-4 and Bases 2.1.1, 3/4.2.2 and 3/4.2.3)

The changes involve (1) decrease of low flow trip setpoint from 91.7% to 90.5%, (2) change of low-low Tavg interlock allowable value from 550.6 to 549.3°F, and

(3) increase of RCS flow uncertainty from 2.0 to 2.1%. The changes are consistent with the assumptions used in the transient analysis to support the reload application and are, therefore, acceptable.

3.9

Core Operating Limits Report (Definitions, 3.1.3.1, 3.1.3.5, 3.1.3.6, Index 3.0/4.0, Figure 3.2-1, 6.9.1)

The licensee proposed to remove cycle specific operating limits from the TS and place them in the cycle specific Core Operating Limits Report (COLR). The operating limits include (1) the control bank insertion limits, and (2) the relaxed axial offset control and base load axial flux difference limits. The related TS changes are: (1) an addition of a defined formal report, the COLR, to the Definition section, (2) modifications of the affected LCOs and Surveillance sections to remove the current references to TS figures and to add references to the COLR, and (3) incorporation of the current Peaking Factor Limit Report section of the administrative controls into the COLR. These changes are consistent with the NRC guidelines specified in NRC Generic Letter 88-16 and are supported by the analytical results provided for the reload application; therefore, the staff concludes that the changes are acceptable.

3.10 Editorial Changes (6.10.2, Bases 3/4.2.2 and 3/4.2.3)

These changes only correct the page headings; therefore, the staff concludes that they are acceptable.

3.11 Summary

The staff has reviewed the licensee's submittal of VANTAGE 5 fuel design and TS changes for the Harris Cycle 3 transition core and all VANTAGE 5 cores. Based on the approved generic topical report WCAP-10444-P-A and plant specific analyses, the staff approves the use of the VANTAGE 5 fuel design, TS changes and the FSAR revisions which incorporate the VANTAGE 5 fuel design for the Harris Cycle 3 transition core and the VANTAGE 5 fuel future cores.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on August 30, 1989 (54 FR 35953). Acccordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant impact on the quality of the human environment.

5.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 29399) on July 12, 1989, and consulted with the State of North Carolina. No public comments or requests for hearing were received, and the State of North Carolina did not have any comments.

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

Principal Contributors: S. Sun
R. Becker
J. Guo

Dated: October 18, 1989

REFERENCES

1. Letter from A. Cutter (CP&L) to NRC, Shearon Harris Nuclear Power Plant, Docket No. 50-400/License No. NPF-63, Request for License Amendment Cycle 3 Reload Submittal, April 17, 1989.

2. Attachment 1 to Reference 1, Safety Evaluation for the Shearon Harris Nuclear Power Plant Transition to Westinghouse 17x17 VANTAGE 5 Fuel.

3. Attachment 2 to Reference 1, Technical Specifications Change Pages for the Shearon Harris Nuclear Power Plant Transition to Westinghouse 17x17 VANTAGE 5 Fuel.

4. Attachment 3 to Reference 1, Non-LOCA Accident Analysis for the Shearon Harris Nuclear Power Plant Transition to Westinghouse 17x17 VANTAGE 5 Fuel.

5. Attachment 4 to Reference 1, LOCA Accident Analysis for the Shearon Harris Nuclear Power Plant Transition to Westinghouse 17x17 VANTAGE 5 Fuel.

6. Attachment 5 to Reference 1, Sample Core Operating Limits Report.

7. Attachment 6 to Reference 1, Significant Hazards Evaluation for the Shearon Harris Nuclear Power Plant Transition to Westinghouse 17x17 VANTAGE 5 Fuel.

8. Letter from A. Cutter (CP&L) to NRC, Responses to NRC Questions on Cycle 3 Technical Specifications Changes Request, October 2, 1989.

9. Letter from A. Cutter (CP&L) to NRC, Supplement to Cycle 3 Reload Amendment Request, June 29, 1989.

10. WCAP-8567, Improved Thermal Design Procedure, July 1975.

11. Letter from NRC to Westinghouse from Stolz to Eicheldinger, SER on WCAP-7956, 8054, 8567 and 8762, April 19, 1978.

12. WCAP-9220-P-A (Proprietory Version) and WCAP-9221-P-A (Non-Proprietory Version), Revision 1, Westinghouse ECCS Evaluation Model - 1981 Version, 1981

13. Letter from J. F. Stoltz (NRC) to T. M. Anderson (Westinghouse), Review of WCAP-8720, Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations.

14. WCAP-8302, SATAN-VI Program: Comprehensive Space Time Dependent Analysis of Loss of Coolant, 1974.

15. WCAP-8170 (Proprietary) and WCAP-8171 (Non-Proprietary), Calculational Model for Core Reflooding After a Loss-of-Coolant Accident - WREFLOOD Code, 1974.

16. WCAP-10266-P-A Rev. 2 (Proprietary), the 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code, 1987.

17. WCAP-8327 (Proprietary) and WCAP-8326 (Non-Proprietary), Containment Pressure Analysis Code (COCO), June 1974.

18. WCAP-10079-P-A, NOTRUMP-A Nodal Transient Small Break and General Network Code, August 1985.

19. WCAP-10054-P-A, Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code, August 1985.

20. WCAP-8301, LOCTA-IV Program: Loss of Coolant Transient Analysis, June 1974.