Catawba Nuclear Station COLR

Catawba Unit 2 Cycle ~~5~~ 6

Core Operating Limits Report

November 22, 1991

Duke Power Company

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Catawba 2 Cycle ~~5~~ 6 Core Operating Limits Report

REVISION LOG

Revision	Effective Date	Effective Pages
Original Issue	22 November 1991	Pages 1 - 15

Catawba 2 Cycle ~~5~~ 6 Core Operating Limits Report

1.0 Core Operating Limits Report

This Core Operating Limits Report, (COLR), for Catawba Unit 2, Cycle ~~5~~ 6 has been prepared in accordance with the requirements of Technical Specification 6.9.1.9.

The Technical Specifications affected by this report are listed below:

3/4.1.1.3	Moderator Temperature Coefficient
3/4.1.3.5	Shutdown Rod Insertion Limit
3/4.1.3.6	Control Rod Insertion Limit
3/4.2.1	Axial Flux Difference
3/4.2.2	Heat Flux Hot Channel Factor
3/4.2.3	Reactor Coolant System Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor

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2.0 Operating Limits

The cycle-specific parameter limits for the specifications listed in section 1.0 are presented in the following subsections. These limits have been developed using NRC-approved methodologies specified in Technical Specification 6.9.1.9.

2.1 Moderator Temperature Coefficient (Specification 3/4.1.1.3)

2.1.1 The Moderator Temperature Coefficient (MTC) Limits are:

The MTC shall be less positive than the limits shown in Figure 1. The BOC/ARO/HZP MTC shall be less positive that $0.7 * 10^{-4}$ $\Delta K/K/^{\circ}F$.

The EOC/ARO/RTP MTC shall be less negative that $-4.1 * 10^{-4}$ $\Delta K/K/^{\circ}F$.

2.1.2 The MTC Surveillance Limit is:

The 300 PPM/ARO/RTP MTC should be less negative than or equal to $-3.2 * 10^{-4}$ $\Delta K/K/^{\circ}F$.

Where: BOC stands for Beginning of Cycle
ARO stands for All Rods Out
HZP stands for Hot Zero (Thermal) Power
EOC stands for End of Cycle
RTP stands for Rated Thermal Power

Catawba 2 Cycle ~~5~~ 6 Core Operating Limits Report

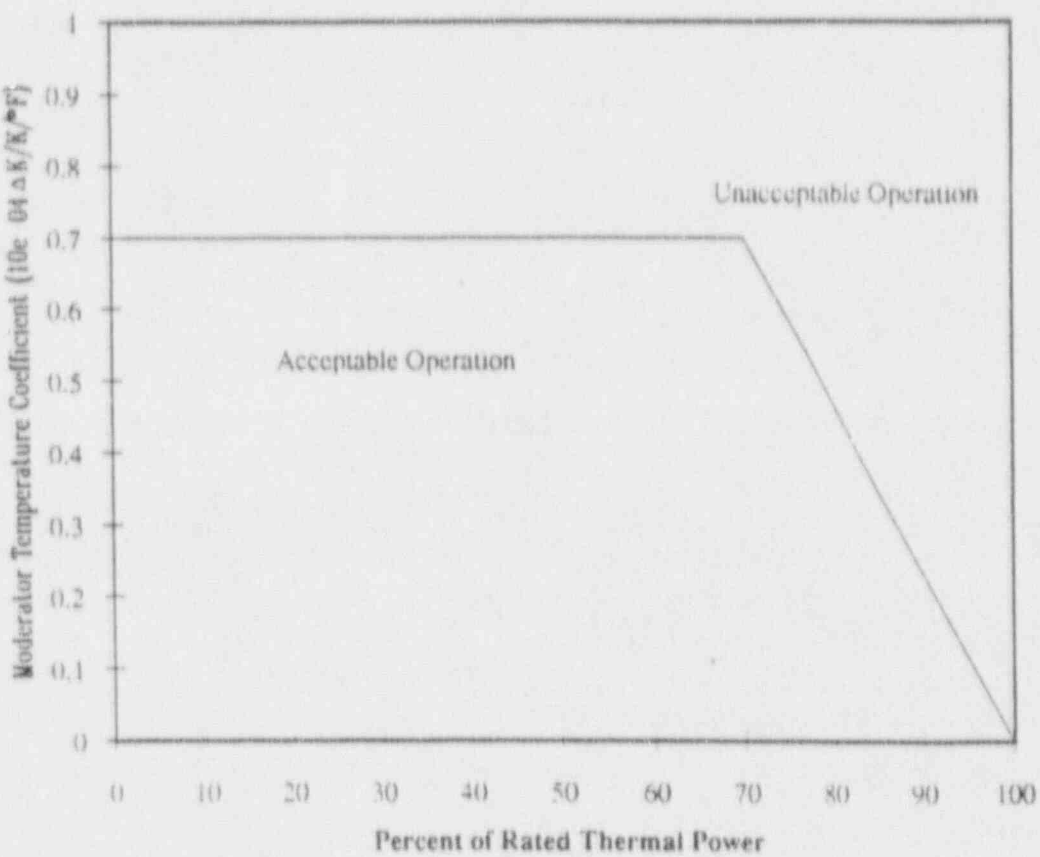

Figure 1

Moderator Temperature Coefficient Versus Power Level

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2.2 Shutdown Rod Insertion Limit (Specification 3/4.1.3.5)

2.2.1 The shutdown rods shall be withdrawn to at least ~~226~~ 222 steps.

2.3 Control Rod Insertion Limits (Specification 3/4.1.3.6)

2.3.1 The control rod banks shall be limited in physical insertion as shown in Figure 2.

2.4 Axial Flux Difference (Specification 3/4.2.1)

~~2.4.1 The AXIAL FLUX DIFFERENCE (AFD) Limits are provided in Figure 3.~~

~~2.4.2 The target band during base load operation is not applicable for Catawba 2 Cycle 5.~~

~~2.4.3 The minimum allowable power level for Base Load Operation (APL^{ND}) is not applicable for Catawba 2 Cycle 5.~~

2.4.1 The Axial Flux Difference (AFD) Limits are provided in Figure 3.

$(\text{AFD Limit})^{\text{COLR}}_{\text{negative}}$ is the negative AFD limit from Figure 3.

$(\text{AFD Limit})^{\text{COLR}}_{\text{positive}}$ is the positive AFD limit from Figure 3.

Catawba 2 Cycle 5 Core Operating Limits Report

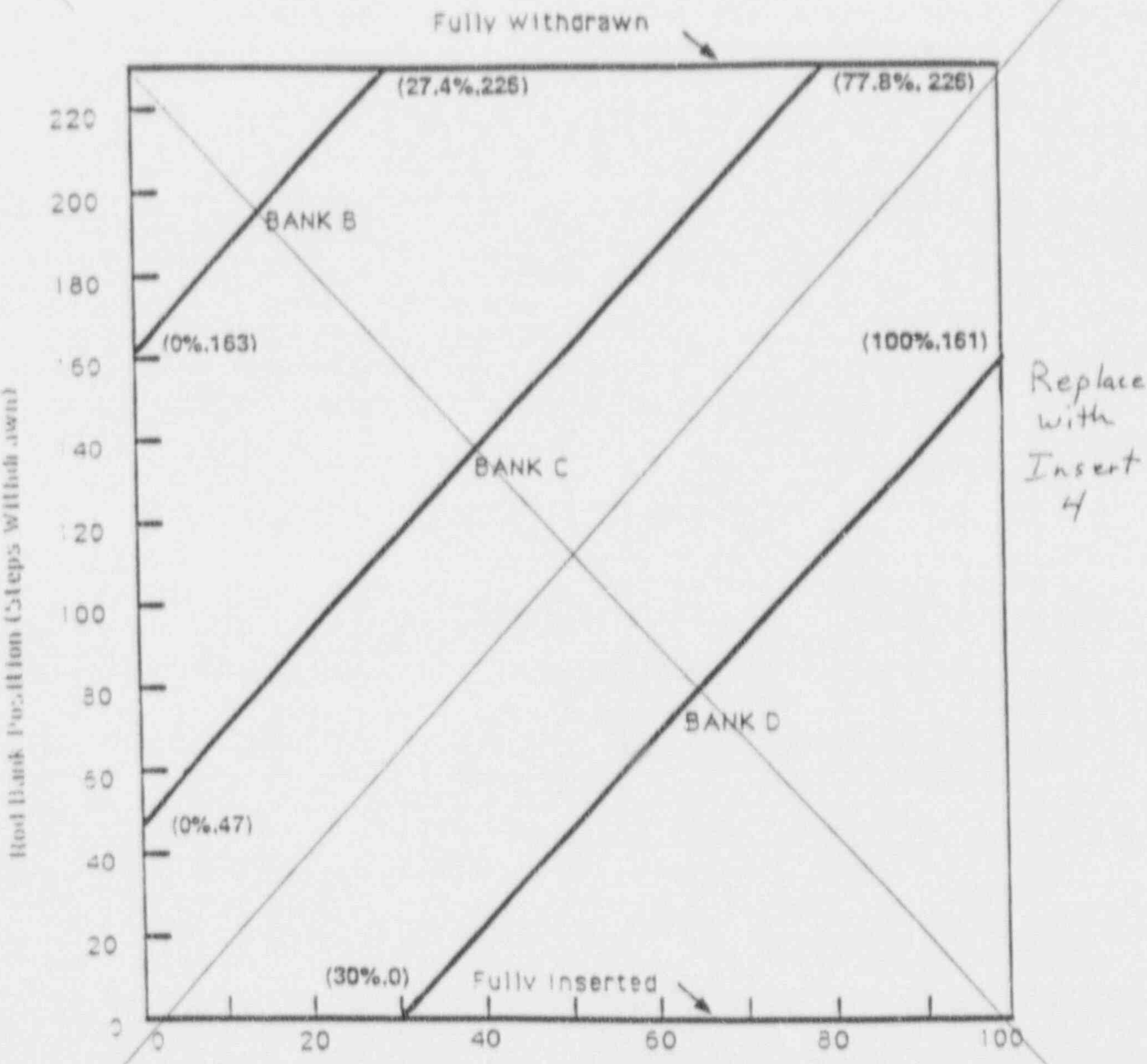

Figure 2

Control Rod Bank Insertion Limits Versus Percent Rated Thermal Power

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Catawba 2 Cycle 6 Core Operating Limits Report

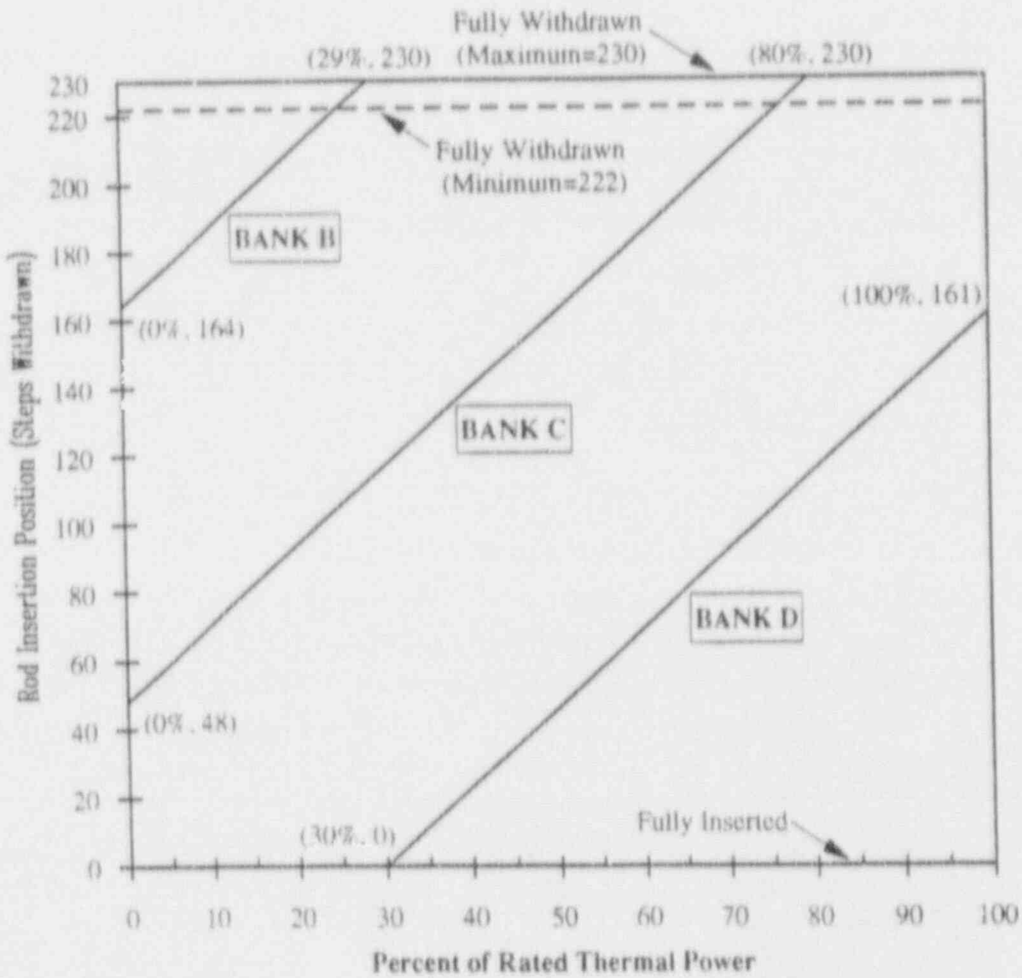

Figure 2

Control Rod Bank Insertion Limits Versus Percent of Rated Thermal Power

Catawba 2 Cycle ~~5~~ 6 Core Operating Limits Report

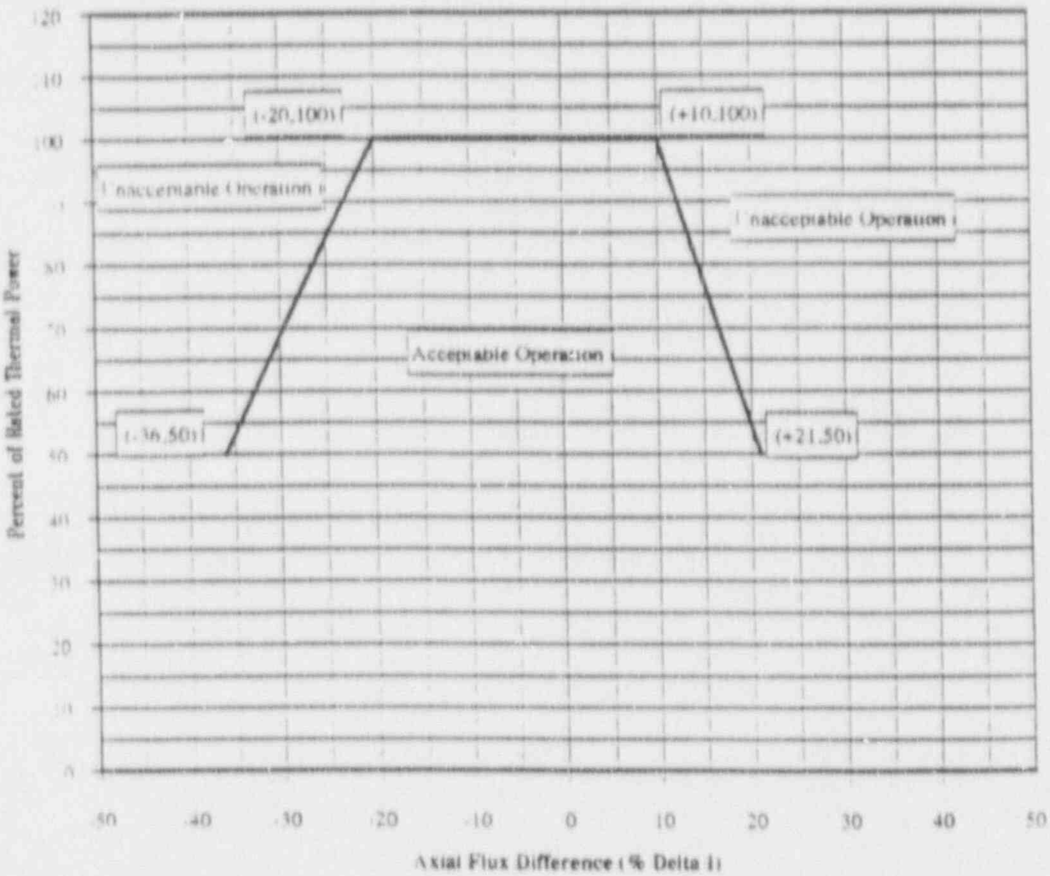

Figure 3

Percent of Rated Thermal Power Versus Axial Flux Difference Limits

Catawba 2 Cycle 5 Core Operating Limits Report

2.5 Heat Flux Hot Channel Factor - $F_Q(Z)$ (Specification 3/4.2.2)

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

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

where: $P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}}$

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2.5.1 $F_Q^{RTP} = 2.32$

2.5.2 K(Z) is provided in Figure 4.

2.5.3 W(Z) values are provided in Figures 5 through 7.

2.5.4 Base load W(Z)'s are not applicable for Catawba 2 Cycle 5.

Insert 5

Catawba 2 Cycle 6 Core Operating Limits Report

2.5 Heat Flux Hot Channel Factor $F_Q(X,Y,Z)$ (Specification 3/4.2.2)

2.5.1 $F_Q^{RTP} = 2.32$

2.5.2 K(Z) is provided in Figure 4 for Mark-BW fuel.

2.5.3 K(Z) is provided in Figure 5 for OFA fuel.

The following parameters are required for core monitoring per the Surveillance Requirements of Specification 3/4.2.2:

2.5.4 $$[F_Q^L(X,Y,Z)]^{OP} = \frac{F_Q^D(X,Y,Z) * M_Q(X,Y,Z)}{UMT * MT * TILT}$$

where $[F_Q^L(X,Y,Z)]^{OP}$ = cycle dependent maximum allowable design peaking factor which ensures that the $F_Q(X,Y,Z)$ limit will be preserved for operation within the LCO limits. $[F_Q^L(X,Y,Z)]^{OP}$ includes allowances for calculational and measurement uncertainties.

$F_Q^D(X,Y,Z)$ = the design power distribution for F_Q. $F_Q^D(X,Y,Z)$ is provided in Table 2 for normal operation and table 2A for power escalation testing during initial startup.

$M_Q(X,Y,Z)$ = the margin remaining in core location X,Y,Z to the LOCA limit in the transient power distribution. $M_Q(X,Y,Z)$ is provided in Table 3 for normal operation and table 3A for power escalation testing during initial startup.

UMT = Measurement Uncertainty (UMT = 1.05).

MT = Engineering Hot Channel Factor (MT = 1.03).

TILT = Peaking penalty that accounts for allowable quadrant power tilt ratio of 1.02.

NOTE: $[F_Q^L(X,Y,Z)]^{OP}$ is the parameter identified as $F_Q^{MAX}(X,Y,Z)$ in DPC-NE-2011PA.

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Catawba 2 Cycle 6 Core Operating Limits Report

2.5.5 $[F_Q^L(X,Y,Z)]^{RPS} = \dfrac{F_Q^D(X,Y,Z) * M_C(X,Y,Z)}{UMT * MT * TILT}$

where $[F_Q^L(X,Y,Z)]^{RPS}$ = cycle dependent maximum allowable design peaking factor which ensures that the centerline fuel melt limit will be preserved for all operation. $[F_Q^L(X,Y,Z)]^{RPS}$ includes allowances for calculational and measurement uncertainties.

$F_Q^D(X,Y,Z)$ = the design power distributions for F_Q. $F_Q^D(X,Y,Z)$ is provided in Table 2 for normal operation and table 2A for power escalation testing during initial startup.

$M_C(X,Y,Z)$ = the margin remaining to the CFM limit in core location X,Y,Z from the transient power distribution. $M_C(X,Y,Z)$ calculations parallel the $M_Q(X,Y,Z)$ calculations described in DPC-NE-2011PA, except that the LOCA limit is replaced with the CFM limit. $M_C(X,Y,Z)$ is provided in Table 4 for normal operation and table 4A for power escalation testing during initial startup.

UMT = Measurement Uncertainty (UMT = 1.05).

MT = Engineering Hot Channel Factor (MT = 1.03).

TILT = Peaking penalty that accounts for allowable quadrant power tilt ratio of 1.02.

NOTE: $[F_Q^L(X,Y,Z)]^{RPS}$ is similar to the parameter identified as $F_Q^{MAX}(X,Y,Z)$ in DPC-NE-2011PA except that $M_C(X,Y,Z)$ replaces $M_Q(X,Y,Z)$.

2.5.6 KSLOPE = 0.078

where KSLOPE = Adjustment to the K_1 value from OTΔT required to compensate for each 1% that $[F_Q^L(X,Y,Z)]^{RPS}$ exceeds it limit

Catawba 2 Cycle 5 Core Operating Limits Report

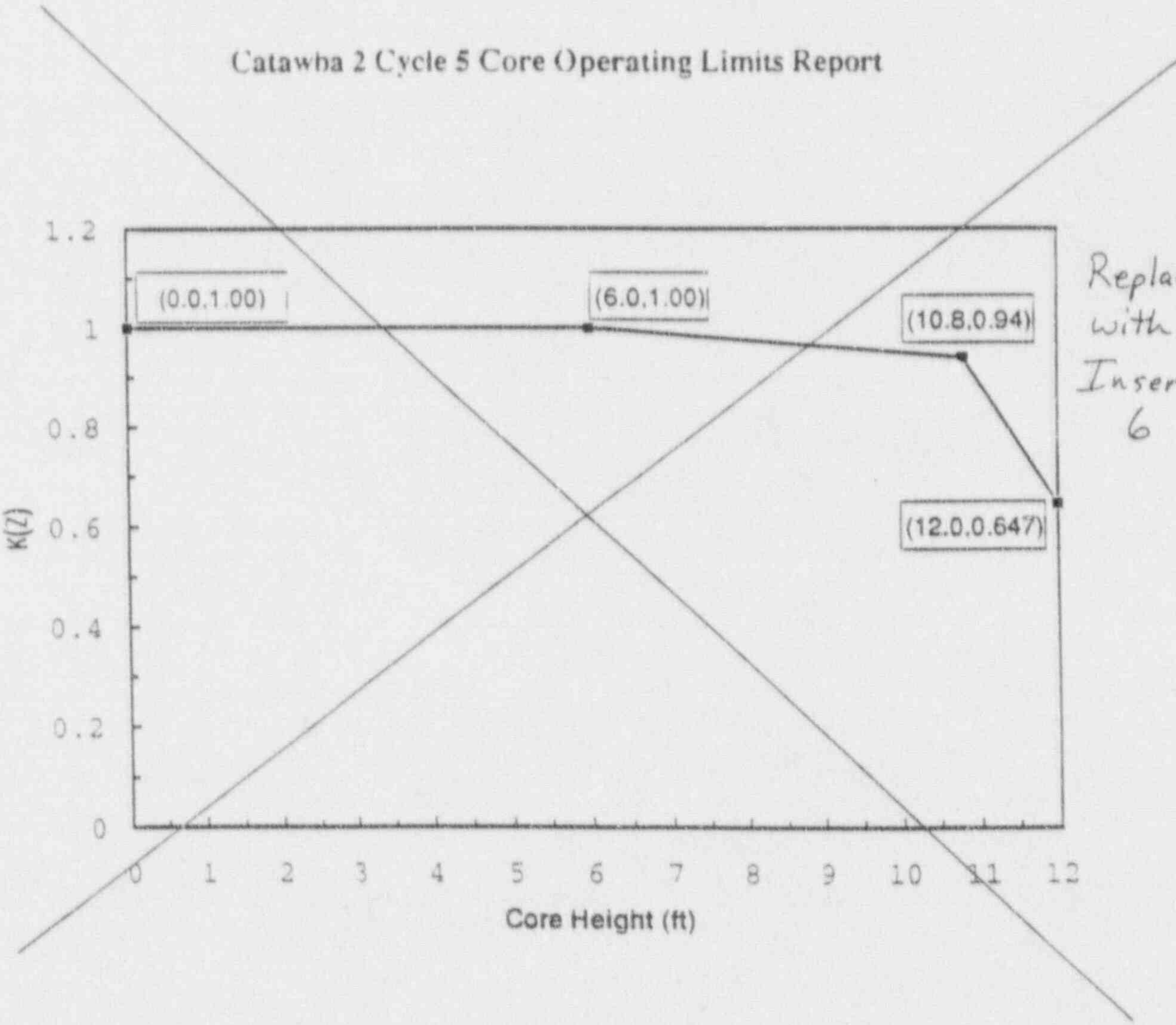

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Figure 4

K(Z), Normalized $F_Q(Z)$ as a Function of Core Height

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Catawba 2 Cycle 6 Core Operating Limits Report

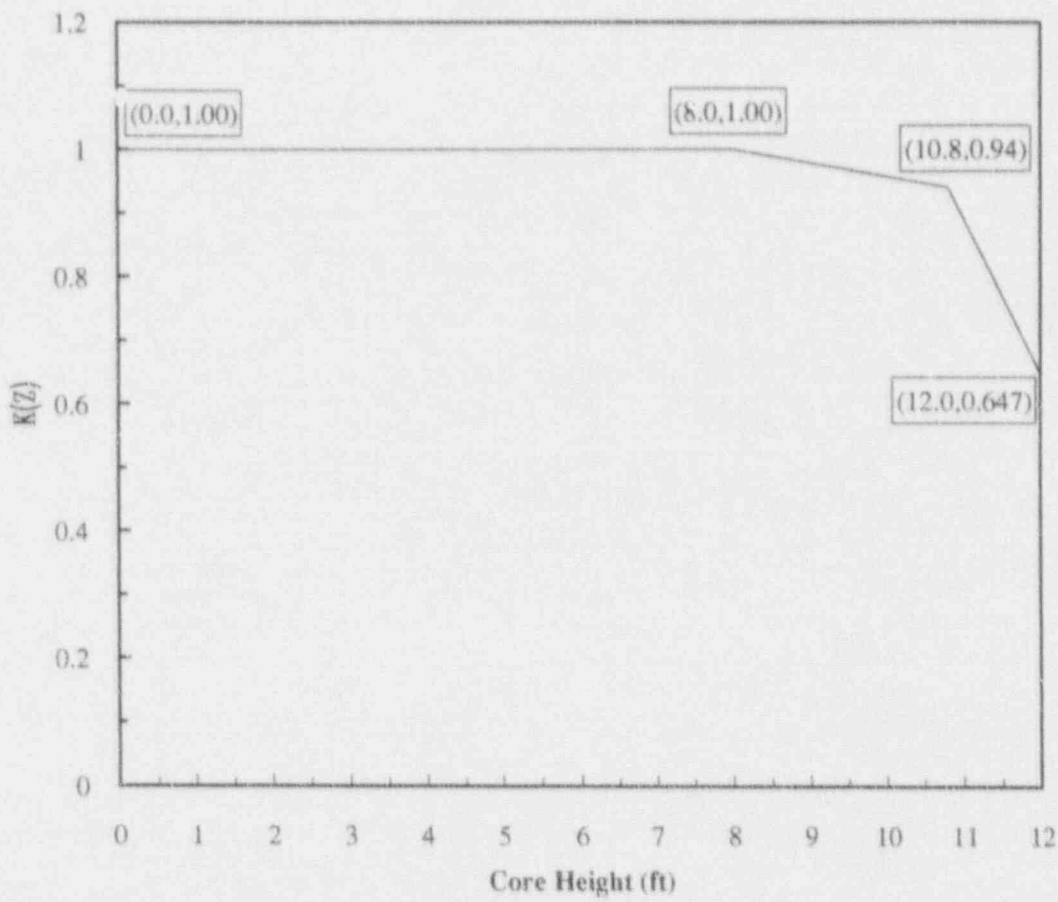


Figure 4

K(Z), Normalized $F_Q(X,Y,Z)$ as a Function of Core Height for Mk-BW Fuel

Insert 6 - continued

Catawba 2 Cycle 6 Core Operating Limits Report

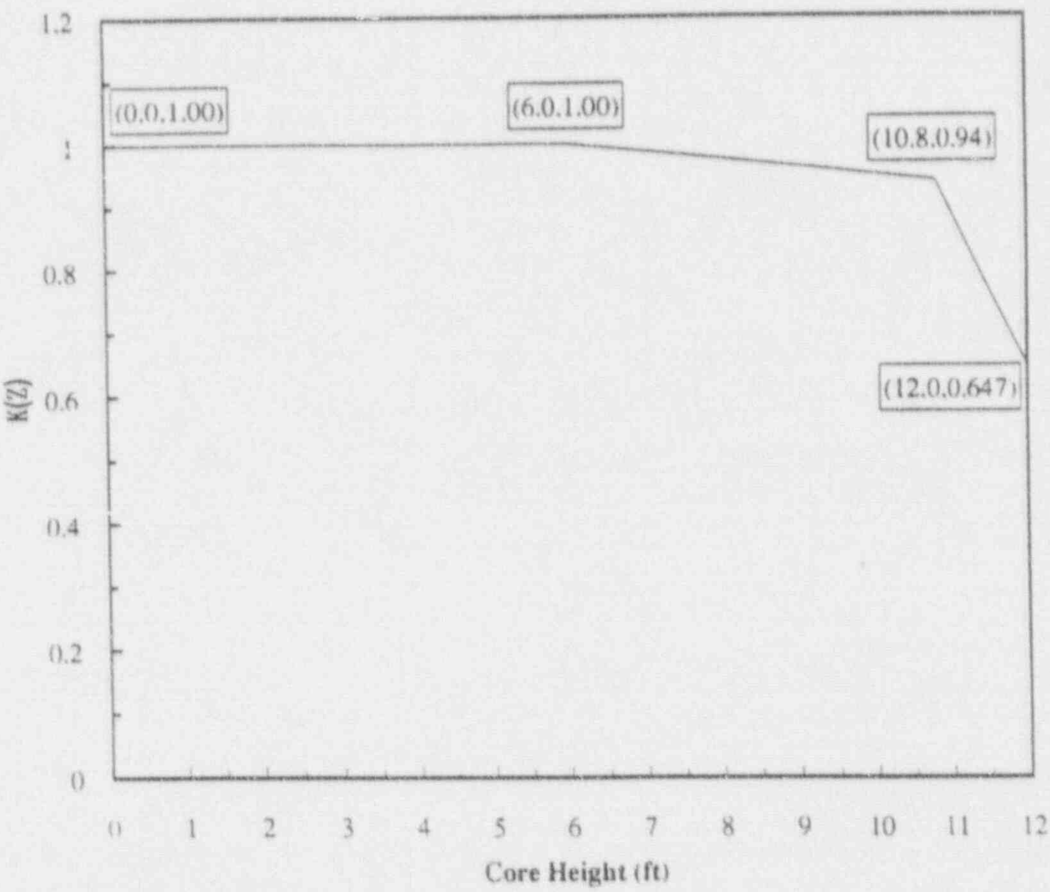
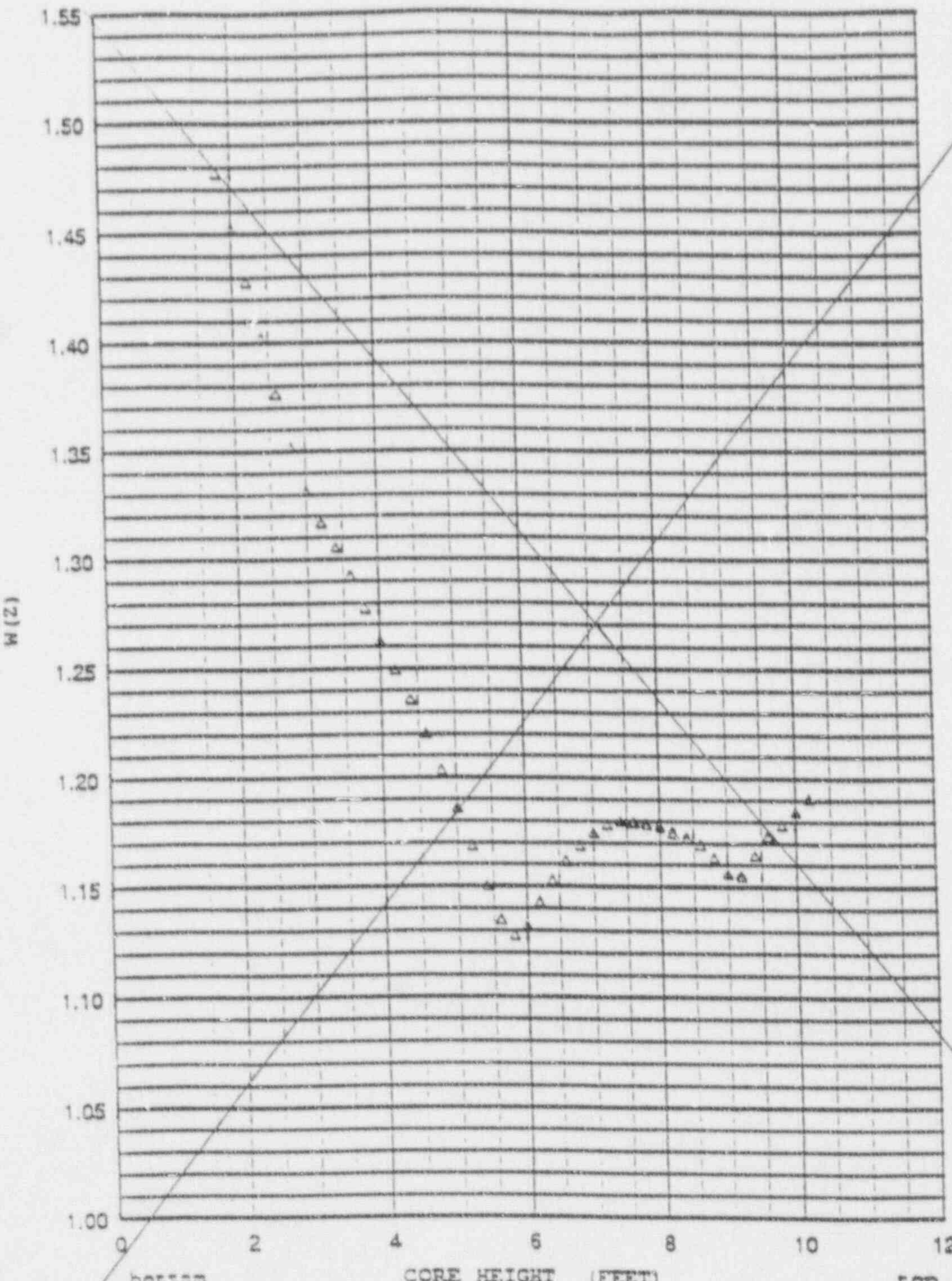


Figure 5

K(Z), Normalized $F_Q(X,Y,Z)$ as a Function of Core Height for OFA Fuel

Catawba 2 Cycle 5 Core Operating Limits Report

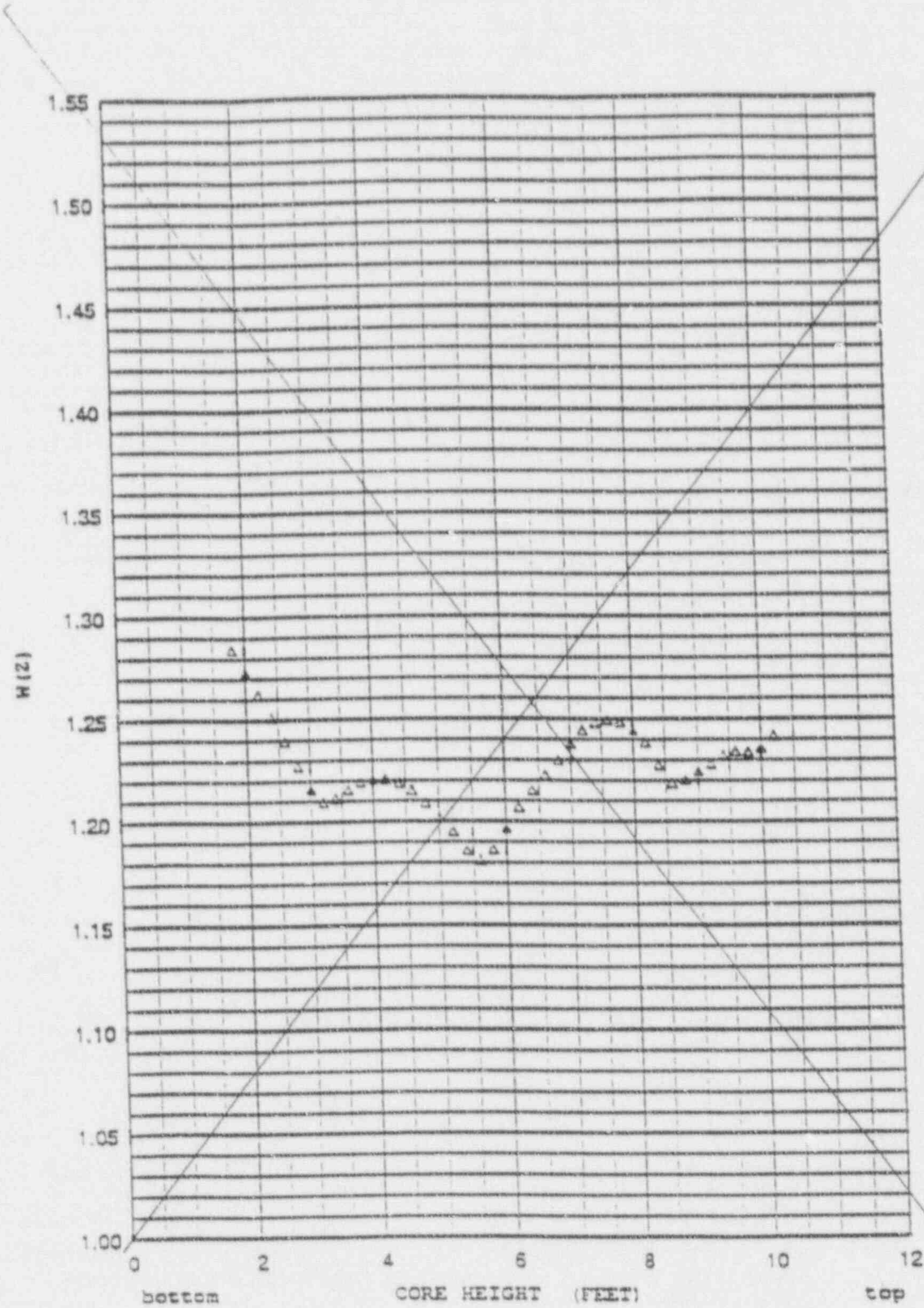


HEIGHT (FEET)	BOL W(Z)
* 0.00	1.0000
* 0.20	1.0000
* 0.40	1.0000
* 0.60	1.0000
* 0.80	1.0000
* 1.00	1.0000
* 1.20	1.0000
* 1.40	1.0000
* 1.60	1.0000
1.80	1.4776
2.00	1.4533
2.20	1.4283
2.40	1.4026
2.60	1.3766
2.80	1.3519
3.00	1.3318
3.20	1.3169
3.40	1.3055
3.60	1.2925
3.80	1.2776
4.00	1.2630
4.20	1.2496
4.40	1.2359
4.60	1.2205
4.80	1.2037
5.00	1.1863
5.20	1.1686
5.40	1.1504
5.60	1.1352
5.80	1.1282
6.00	1.1325
6.20	1.1432
6.40	1.1530
6.60	1.1617
6.80	1.1690
7.00	1.1747
7.20	1.1787
7.40	1.1804
7.60	1.1796
7.80	1.1787
8.00	1.1775
8.20	1.1747
8.40	1.1723
8.60	1.1694
8.80	1.1637
9.00	1.1565
9.20	1.1550
9.40	1.1647
9.60	1.1734
9.80	1.1787
10.00	1.1848
10.20	1.1909
* 10.40	1.0000
* 10.60	1.0000
* 10.80	1.0000
* 11.00	1.0000
* 11.20	1.0000
* 11.40	1.0000
* 11.60	1.0000
* 11.80	1.0000
* 12.00	1.0000

Figure 5
Catawba Unit 2 Cycle 5
RAOC W(Z) at 150 MWD/MTU

Top and Bottom 15% excluded as per Tech. Spec. 4.2.2.2.G

Catawba 2 Cycle 5 Core Operating Limits Report

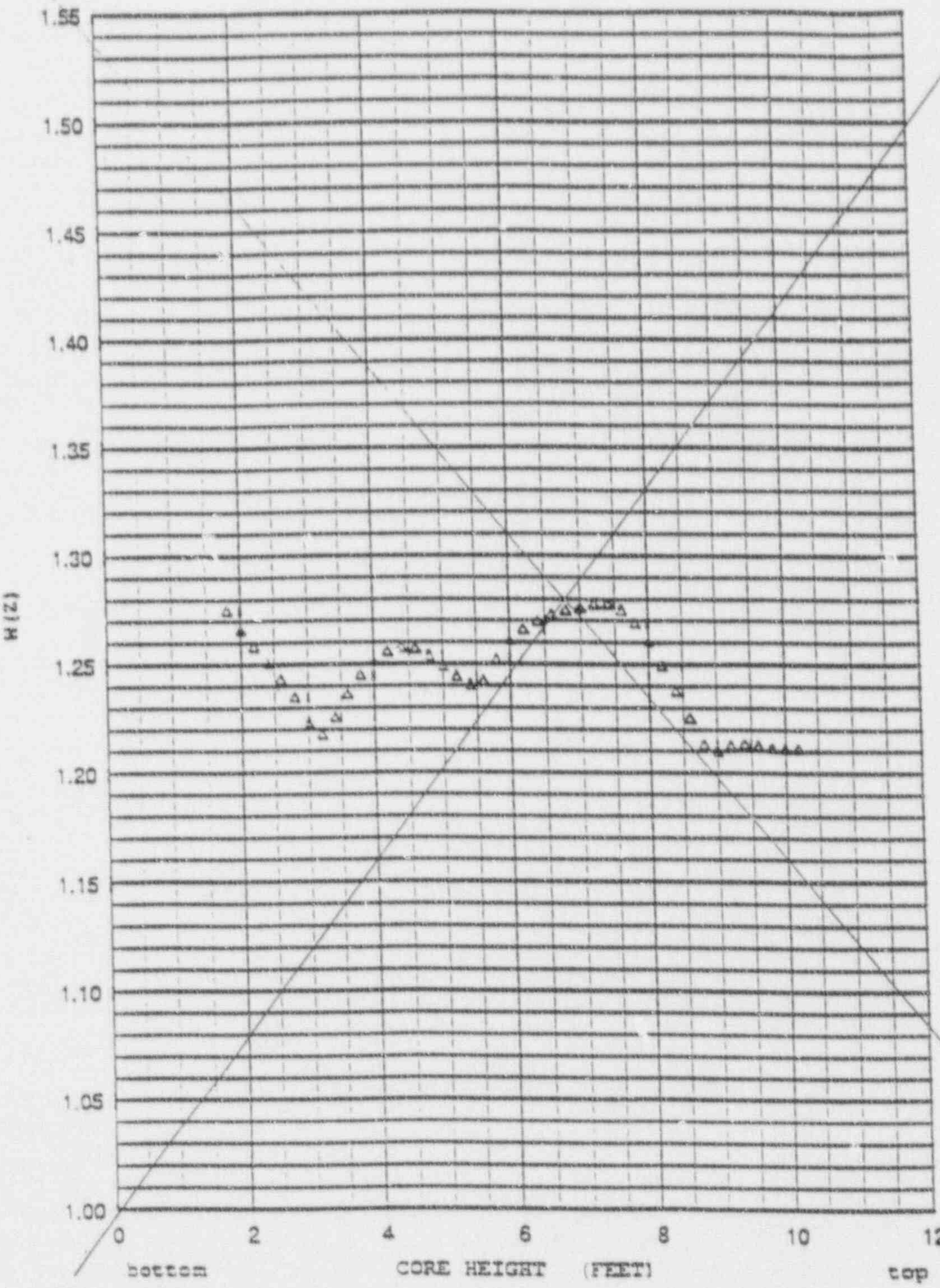


HEIGHT (FEET)	MOL W(Z)
* 0.00	1.0000
* 0.20	1.0000
* 0.40	1.0000
* 0.60	1.0000
* 0.80	1.0000
* 1.00	1.0000
* 1.20	1.0000
* 1.40	1.0000
* 1.60	1.0000
1.80	1.2844
2.00	1.2735
2.20	1.2624
2.40	1.2512
2.60	1.2396
2.80	1.2274
3.00	1.2159
3.20	1.2097
3.40	1.2114
3.60	1.2157
3.80	1.2193
4.00	1.2209
4.20	1.2207
4.40	1.2188
4.60	1.2150
4.80	1.2091
5.00	1.2022
5.20	1.1951
5.40	1.1861
5.60	1.1812
5.80	1.1866
6.00	1.1968
6.20	1.2070
6.40	1.2153
6.60	1.2226
6.80	1.2301
7.00	1.2382
7.20	1.2444
7.40	1.2482
7.60	1.2496
7.80	1.2484
8.00	1.2448
8.20	1.2387
8.40	1.2275
8.60	1.2187
8.80	1.2205
9.00	1.2245
9.20	1.2280
9.40	1.2325
9.60	1.2340
9.80	1.2337
10.00	1.2359
10.20	1.2419
* 10.40	1.0000
* 10.60	1.0000
* 10.80	1.0000
* 11.00	1.0000
* 11.20	1.0000
* 11.40	1.0000
* 11.60	1.0000
* 11.80	1.0000
* 12.00	1.0000

Figure 6
Catawba Unit 2 Cycle 5
RAOC W(Z) at 5000 MWD/MTU

Top and Bottom 15% excluded as per Tech. Spec. 4.2.2.2.G

Catawba 2 Cycle 5 Core Operating Limits Report



HEIGHT (FEET)	EOL W(Z)
* 0.00	1.0000
* 0.20	1.0000
* 0.40	1.0000
* 0.60	1.0000
* 0.80	1.0000
* 1.00	1.0000
* 1.20	1.0000
* 1.40	1.0000
* 1.60	1.0000
1.80	1.2752
2.00	1.2663
2.20	1.2586
2.40	1.2509
2.60	1.2433
2.80	1.2352
3.00	1.2235
3.20	1.2177
3.40	1.2259
3.60	1.2365
3.80	1.2451
4.00	1.2517
4.20	1.2560
4.40	1.2579
4.60	1.2573
4.80	1.2540
5.00	1.2492
5.20	1.2444
5.40	1.2405
5.60	1.2423
5.80	1.2525
6.00	1.2612
6.20	1.2664
6.40	1.2703
6.60	1.2731
6.80	1.2752
7.00	1.2766
7.20	1.2780
7.40	1.2780
7.60	1.2750
7.80	1.2693
8.00	1.2605
8.20	1.249
8.40	1.238
8.60	1.225
8.80	1.213
9.00	1.210
9.20	1.213
9.40	1.213
9.60	1.212
9.80	1.211
10.00	1.211
10.20	1.211
* 10.40	1.000
* 10.60	1.000
* 10.80	1.000
* 11.00	1.000
* 11.20	1.000
* 11.40	1.000
* 11.60	1.000
* 11.80	1.000
* 12.00	1.000

Figure 7
Catawba Unit 2 Cycle 5
RAOC W(Z) at 13000 MWD/MTU

Top and Bottom 15% excluded as per Tech. Spec. 4.2.2.2.G

Catawba 2 Cycle 5 Core Operating Limits Report

2.6 ~~RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor~~ - $F^N_{\Delta H}$
(Specification 3/4.2.3)

$$R = \frac{F^N_{\Delta H}}{F^{RTP}_{\Delta H} * (1 + MF_{\Delta H} * (1-P))}$$

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where: $P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}}$

2.6.1 $F^{RTP}_{\Delta H} = 1.49$

2.6.2 $MF_{\Delta H} = 0.3$

2.6.3 The Acceptable Operation Region from the combination of Reactor Coolant System total flow and R is provided in Figure 8.

Insert 7

Catawba 2 Cycle 6 Core Operating Limits Report

2.6 Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}(X,Y,Z)$ (Specification 3/4.2.3)

The following parameters are required for core monitoring per the LCO Requirements of Specification 3/4.2.3:

2.6.1 $$[F_{\Delta H}(X,Y)]^{LCO} = MARP\ (X,Y) * \left[1.0 + \frac{1}{RRH} * (1.0 - P)\right]$$

where $(MARP(X,Y))$ = Catawba 2 Cycle 6 Operating Limit Maximum Allowable Radial Peaks. $(MARP(X,Y))$ is provided in Table 1.

$$P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}}$$

The following parameters are required for core monitoring per the Surveillance Requirements of Specification 3/4.2.3:

2.6.2 $$[F^{L}_{\Delta H}(X,Y)]^{SURV} = \frac{F^{D}_{\Delta H}(X,Y) * M_{\Delta H}(X,Y)}{UMR * TILT}$$

where $[F^{L}_{\Delta H}(X,Y)]^{SURV}$ = cycle dependent maximum allowable design peaking factor which ensures that the $F_{\Delta H}(X,Y)$ limit will be preserved for operation within the LCO limits. $[F^{L}_{\Delta H}(X,Y)]^{SURV}$ includes allowances for calculational and measurement uncertainties.

$F^{D}_{\Delta H}(X,Y)$ = the design power distribution for $F_{\Delta H}$. $F^{D}_{\Delta H}(X,Y)$ is provided in Table 5 for normal operation and table 5A for power escalation testing during initial startup.

$M_{\Delta H}(X,Y)$ = the margin remaining in core location X,Y to the Operational DNB limit in the transient power distribution. $M_{\Delta H}(X,Y)$ is provided in Table 6 for normal operation and table 6A for power escalation testing during initial startup.

UMR = Uncertainty value for measured radial peaks, (UMR = 1.04).

TILT = Peaking penalty that accounts for allowable quadrant power tilt ratio of 1.02.

Catawba 2 Cycle 6 Core Operating Limits Report

NOTE: $[F^{L}_{\Delta H}(X,Y)]^{SURV}$ is the parameter identified as $F^{MAX}_{\Delta H}(X,Y)$ in DPC-NE-2011PA.

2.6.3 RRH = 3.34

where RRH = Thermal Power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds its limit.

2.6.4 TRH = 0.04

where TRH = Reduction in OTΔT K_1 setpoint required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds its limit.

Insert 7 - continued **Catawba 2 Cycle 6 Core Operating Limits Report**

Table 1. Maximum Allowable Radial Peaks (MARPs)

Core Height (ft)	1.1 Axial Peak MARP	1.2 Axial Peak MARP	1.3 Axial Peak MARP	1.4 Axial Peak MARP
0.12	1.5809	1.6266	1.6722	1.7113
1.2	1.5806	1.6259	1.6677	1.7085
2.4	1.5836	1.6265	1.6663	1.7025
3.6	1.5859	1.6263	1.6635	1.6960
4.8	1.5871	1.6240	1.6571	1.6751
6.0	1.5878	1.6196	1.6470	1.6303
7.2	1.5864	1.6130	1.6265	1.5848
8.4	1.5781	1.5956	1.5773	1.5327
9.6	1.5655	1.5612	1.5208	1.4815
10.8	1.5459	1.5152	1.4717	1.4292
12.0	1.5133	1.4693	1.4274	1.3878

Core Height (ft)	1.5 Axial Peak MARP	1.6 Axial Peak MARP	1.7 Axial Peak MARP	1.8 Axial Peak MARP
0.12	1.7477	1.7331	1.7054	1.6438
1.2	1.7463	1.7029	1.6789	1.6193
2.4	1.7126	1.6616	1.6433	1.5869
3.6	1.6735	1.6211	1.6011	1.5504
4.8	1.6313	1.5811	1.5622	1.5121
6.0	1.5868	1.5415	1.5238	1.4763
7.2	1.5378	1.4913	1.4766	1.4344
8.4	1.4886	1.4450	1.4296	1.3880
9.6	1.4399	1.4013	1.3882	1.3490
10.8	1.3883	1.3526	1.3433	1.3081
12.0	1.3500	1.3140	1.3078	1.2749

Core Height (ft)	1.9 Axial Peak MARP	2.1 Axial Peak MARP
0.12	1.5839	1.5401
1.2	1.5624	1.5154
2.4	1.5328	1.4801
3.6	1.5013	1.4395
4.8	1.4626	1.4030
6.0	1.4291	1.3619
7.2	1.3920	1.3271
8.4	1.3485	1.2824
9.6	1.3126	1.2501
10.8	1.2726	1.2091
12.0	1.2443	1.1890

Catawba 2 Cycle 5 Core Operating Limits Report

Penalties of 0.1% for undetected feedwater venturi fouling and measurement uncertainties of 2.1% for flow and 4.0% for incore measurement of $F^N_{\Delta H}$ are included in this figure.

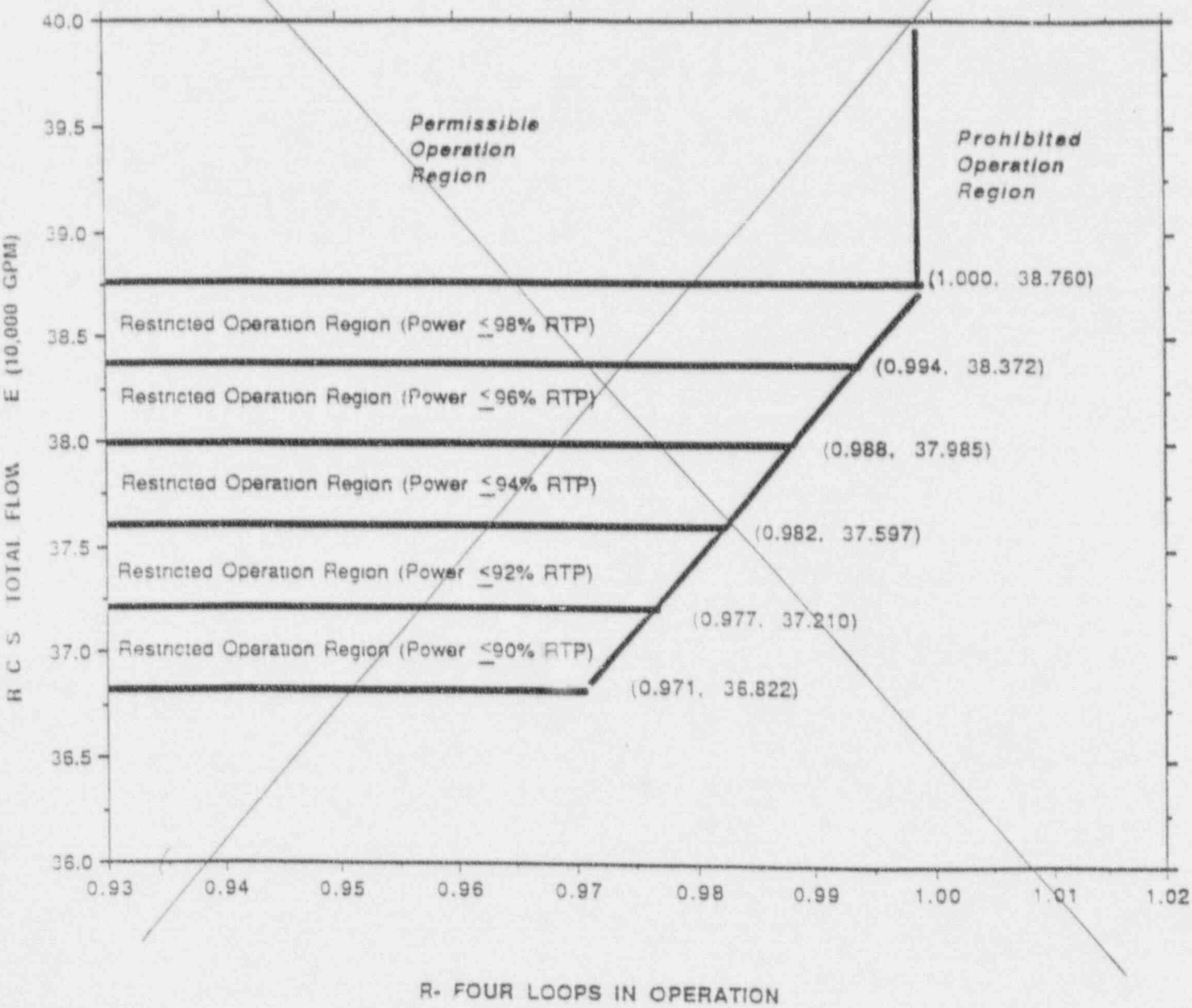

Figure 8
RCS Flow vs. R-Four Loops in Operation

8.3 Changes to the Catawba FSAR

Notes to Table 6-77

Containment Isolation Valve and Actuation Data

Notes:

1. Valve arrangements are shown in Figure 6-112.
2. Definition of Actuation Signals

   S - Safety Injection Signal (T signal also activated by S signal)

   T - Containment Isolation Signal (Phase A containment isolation)

   P - Containment High-High Pressure Signal (Phase B containment isolation)

0 3. Valve Type Abbreviations

0 GL Globe

causes main steamline isolation and

0 SW Swing Check

0 GT Gate

0 CK Check

0 RV Relief

0 DS Double Seal

0 FG Flange

0 PG Plug

0 BF Butterfly

0 DP Diaphragm

0 SV Safety

0 SC Stop Check

4. Symbols:

   Valve Position Abbreviations

   O Open

   C Closed

   A Automatic

   R Remote Operation

   M Manual Local Operation

   LC Locked Closed

   C/O Closed prior to Sump or Hot Leg Recirculation: Open after Sump or Hot Leg Recirculation

   LO Locked Open

   AI Fails As is

   Actuator Type

   E Motor (Power Source - Electricity)

1 Dilution Flow Rate

maximum
1 In the absence of flow rate restrictions, the dilution flow rate assumed to enter the RCS is greater than or
1 equal to the ~~design~~ volumetric flow rate of both reactor makeup water pumps. In a dilution event, these
1 pumps are assumed to deliver unborated water to the suction of the centrifugal charging pumps. Since
1 the water delivered by these pumps is typically colder than the RCS inventory, the unborated water
1 expands within the RCS, causing a given volumetric flow rate measured at the colder temperature to
1 correspond to a larger volumetric dilution flow rate within the RCS. This density difference in the
1 dilution flow rate is accounted for in the analysis.

1 Results

1 The calculated sequence of events is shown in Table 15-23.

1 Dilution During Modes in which the BDMS is Required (Modes 3-6)

1 During Mode 6 an inadvertent dilution from the Reactor Makeup Water System is prevented by
1 administrative controls which isolate the RCS from potential sources of unborated makeup water. The
1 results presented in Table 15-23 for this mode are for an assumed dilution event, for which no mechanism
1 or flow path has been identified. For Modes 3-6 with the BDMS operable, the results presented in
1 Table 15-23 show that there is adequate time to reach the BDMS alarm setpoint, stroke closed the valves
1 to isolate the source of unborated water, and purge the unborated water already in the CVCS piping,
1 before the shutdown margin is exhausted. For Modes 3-6 with the BDMS inoperable, the results
1 presented in Table 15-23 show that, with limitations on flow rates from potential sources of unborated
1 water, there is adequate time for the operator to determine the cause of the dilution, isolate the source of
1 unborated water, and initiate reboration before the shutdown margin is exhausted. In accordance with
1 Reference 11, adequate time is judged to be at least 15 minutes for Modes 3-5 and at least 30 minutes for
1 Mode 6. The results presented in Table 15-23 are for the dilution flow rates which, assuming the boron
1 concentration ratios are at the reload safety analysis limits, give exactly these operator response times.
1 Flow rates are restricted, through Technical Specifications and administrative controls, to values which are
1 less than these analyzed flow rates, thus in practice giving even longer operator response times. Additional
1 margin is provided by the fact there is typically margin between the assumed boron concentration ratio for
1 a given mode and the actual corresponding concentration ratio for the reload core.

Dilution During Startup (Mode 2)

This mode of operation is a transitory mode to go to power and is the operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual control with the operator required to maintain a very high awareness of the plant status. For a normal approach to criticality the operator must manually initiate a limited dilution and subsequently manually withdraw the control rods, a process that takes several hours. The plant Technical Specifications require that the operator determine the estimated critical position of the control rods prior to approaching criticality thus assuring that the reactor does not go critical with rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the Source Range reactor trip after receiving P-6 from the Intermediate Range (nominally at 10^5 cps). To fast a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the Source Range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor, allowing sufficient time prior to a loss of shutdown margin for the operator to terminate the dilution event.

However, in the event of an unplanned approach or dilution during power escalation while in the startup mode, the plant status is such that minimal impact will result. The plant will slowly escalate in power to

Table 15-23 (Page 3 of 4). Time Sequence of Events for Incidents Which Cause Reactivity and Power Distribution Anomalies

	Accident	Event	Time (sec.)
1	4b. Dilution during hot shutdown (BDMS inoperable)	Dilution begins	0
1		High-flux-at-shutdown alarm setpoint reached	1816
1		Operator terminates dilution	< 2716
1	5a. Dilution during cold shutdown (BDMS operable)	Dilution begins	0
1		BDMS setpoint reached	~~739~~ 806
1		Dilution source isolated	~~764~~ 831
1		Borated water reaches core	~~≤885~~ < 971
1	5b. Dilution during cold shutdown (BDMS inoperable)	Dilution begins	0
1		High-flux-at-shutdown alarm setpoint reached	~~1816~~ 1820
1		Operator terminates dilution	~~≤2716~~ < 2720
1	6a. Dilution during refueling (BDMS operable)	Dilution begins	0
1		BDMS setpoint reached	1024
1		Dilution source isolated	1049
1		Borated water reaches core	< 1267
1	6b. Dilution during refueling (BDMS inoperable)	Dilution begins	0
1		High-flux-at-shutdown alarm setpoint reached	3441
1		Operator terminates dilution	< 5241
	Rod Cluster Control Assembly Ejection		
	1. Beginning of Life, Full Power	Initiation of rod ejection	0.0
		Power range high neutron flux high setpoint reached	0.05
		Peak nuclear power occurs	0.14
		Rods begin to fall into core	0.55
		Peak fuel average temperature occurs	2.36

9. REFERENCES

1. Catawba Nuclear Station, Final Safety Analysis Report, Docket Nos. 50 - 413/414.

2. BAW-10172P-A, Mark-BW Mechanical Design Report, Babcock & Wilcox, Lynchburg, Virginia, December 19, 1989.

3. DPC-NE-2001P-A, Rev. 1, Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel, Duke Power Company, October 1990.

4. BAW-10084A, Rev. 2, Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, Babcock & Wilcox, October 1978.

5. BAW-10141P-A, Rev. 1, TACO2 - Fuel Performance Analysis, Babcock & Wilcox, June 1983.

6. DPC-NF-2010A, McGuire Nuclear Station/Catawba Nuclear Station Nuclear Physics Methodology for Reload Design, Duke Power Company, June 1985.

7. DPC-NE-2011P-A, Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors, Duke Power Company, March 1990.

8. DPC-NE-2004P-A, McGuire and Catawba Nuclear Stations Core Thermal- Hydraulic Methodology using VIPRE-01, Duke Power Company, December 1991.

9. BAW-10159P-A, BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies, Babcock & Wilcox, July 1990.

10. BAW-10173P-A, Mark-BW Reload Safety Analysis for Catawba and McGuire, Babcock & Wilcox, Revision 2, February 20, 1991.

11. DPC-NE-3000P, Duke Power Company, Thermal-Hydraulic Transient Analysis Methodology, Revision 2, February 20, 1990.

12. DPC-NE-3001-PA, Duke Power Company, Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, November 1991.

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

14. BAW-10168-A, B&W Loss-of-Coolant Accident Evaluation Model For Recirculating Steam Generator Plants, Babcock & Wilcox, Lynchburgh, Virginia, January 1991.

15. DPC-NE-1003A, Revision 1, McGuire Nuclear Station/Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testing, December 1986.

9. REFERENCES

1. Catawba Nuclear Station, Final Safety Analysis Report, Docket Nos. 50 - 413/414.

2. BAW-10172P-A, Mark-BW Mechanical Design Report, Babcock & Wilcox, Lynchburg, Virginia, December 19, 1989.

3. DPC-NE-2001P-A, Rev. 1, Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel, Duke Power Company, October 1990.

4. BAW-10084A, Rev. 2, Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, Babcock & Wilcox, October 1978.

5. BAW-10141P-A, Rev. 1, TACO2 - Fuel Performance Analysis, Babcock & Wilcox, June 1983.

6. DPC-NF-2010A, McGuire Nuclear Station/Catawba Nuclear Station Nuclear Physics Methodology for Reload Design, Duke Power Company, June 1985.

7. DPC-NE-2011P-A, Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors, Duke Power Company, March 1990.

8. DPC-NE-2004P-A, McGuire and Catawba Nuclear Stations Core Thermal- Hydraulic Methodology using VIPRE-01, Duke Power Company, December 1991.

9. BAW-10159P-A, BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies, Babcock & Wilcox, July 1990.

10. BAW-10173P-A, Mark-BW Reload Safety Analysis for Catawba and McGuire, Babcock & Wilcox, Revision 2, February 20, 1991.

11. DPC-NE-3000P, Duke Power Company, Thermal-Hydraulic Transient Analysis Methodology, Revision 2, February 20, 1990.

12. DPC-NE-3001-PA, Duke Power Company, Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, Revision 2, November 1991.

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

14. BAW-10168-A, B&W Loss-of-Coolant Accident Evaluation Model For Recirculating Steam Generator Plants, Babcock & Wilcox, Lynchburgh, Virginia, January 1991.

15. DPC-NE-1003A, Revision 1, McGuire Nuclear Station/Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testing, December 1986.

16. DPC-NE-3002-A, McGuire Nuclear Station/Catawba Nuclear Station FSAR Chapter 15 System Transient Analysis Methodology, November 1991.

17. McGuire Nuclear Station Unit 1, Docket Number 50-369, Cycle 8 Reload Submittal, Duke Power Company, June 26,1991.

18. McGuire Nuclear Station Unit 2, Docket Number 50-370, Cycle 8 Reload Submittal, Duke Power Company, December 18,1991.

19. Catawba Nuclear Station Unit 1, Docket Numbers 50-413 and 50-414, Cycle 7 Reload Submittal, Duke Power Company, April, 13, 1992.

20. Catawba Nuclear Station Unit 1, Docket Numbers 50-413 and 50-414, Cycle 6 Reload Submittal, Duke Power Company, January 9, 1991.

16. DPC-NE-3002-A, McGuire Nuclear Station/Catawba Nuclear Station FSAR Chapter 15 System Transient Analysis Methodology, Revision 1, November 1991.

17. McGuire Nuclear Station Unit 1, Docket Number 50-369, Cycle 8 Reload Submittal, Duke Power Company, June 26,1991.

18. McGuire Nuclear Station Unit 2, Docket Number 50-370, Cycle 8 Reload Submittal, Duke Power Company, December 18,1991.

19. Catawba Nuclear Station Unit 1, Docket Numbers 50-413 and 50-414, Cycle 7 Reload Submittal, Duke Power Company, April, 13, 1992.

20. Catawba Nuclear Station Unit 1, Docket Numbers 50-413 and 50-414, Cycle 6 Reload Submittal, Duke Power Company, January 9, 1991.

Attachment 2

Technical Justification

The Technical Specification (TS) and COLR changes as noted in Tables 8-1 and 8-2 are identical to those previously submitted and approved for Catawba Unit 1 Cycle 7. These changes reflect the transition from Westinghouse to B&W supplied fuel and to Duke analysis methodology. Recent cycles where changes were submitted and approved involving this transition are Catawba Unit 1 Cycles 6 & 7, McGuire Unit 1 Cycle 8 and McGuire Unit 2 Cycle 8. The three exceptions to these changes, as denoted by an asterisk in Table 8-1, are new changes which have not been previously submitted.

Proposed Revision to Technical Specification 2.1.1 & Figure 2.1-1b

This proposed Technical Specification revision deletes Figure 2.1-1b and uses the current Figure 2.1-1a to reflect use of the BWCMV CHF correlation and Duke Power Company's Statistical Core Design (SCD) methodology with a 1.55 thermal design DNBR limit.

Technical Justification

These proposed revisions are the same as those approved for Catawba 1 Cycle 7 (Reference 9).

Proposed Revision to Technical Specification Table 2.2-1

This proposed Technical Specification revision changes the K values for the overtemperature and overpower ΔT trip functions to reflect the use of the BWCMV CHF correlation and Duke Power Company's Statistical Core Design (SCD) methodology with a 1.55 thermal design DNBR limit. In addition, an axial imbalance penalty, $f_2(\Delta I)$, is applied to the OPΔT reactor trip. The power range neutron flux negative rate reactor trip is deleted from the Reactor Protection System.

Technical Justification

These proposed revisions are the same as those approved for Catawba 1 Cycle 7 (Reference 9).

Proposed Revision to Technical Specification 3/4.2.1

This proposed revision provides Axial Flux Difference (AFD) limits consistent with Duke Power Company methodology.

Technical Justification

The proposed revisions are the same as the Axial Flux Difference (AFD) limit changes in the approved submittal for Catawba 1 Cycle 6 (Reference 10).

Proposed Revision to Technical Specification 3/4.2.2

Specification 3/4.2.2 was revised to reflect the power peaking surveillance method described in DPC-NE-2011PA. These revisions are summarized as follows:

1. The statement of the LCO was revised to reflect new nomenclature for the heat flux hot channel factor [$(F_Q(X,Y,Z)$] required by the

methodology in DPC-NE-2011PA and used throughout the Reload Report. Also, as discussed in the McGuire 2 Cycle 8 reload submittal (Reference 8), separate K(Z) curves are provided for the Mark-BW and OFA fuel types.

2. Action a in the current specification has been replaced by Actions a, b, and c in the new specification. The thermal power reduction required when $F_Q(X,Y,Z)$ exceeds its limit are the same as the current requirement, as is the reduction required in the OPΔT trip setpoints. Action b is a new requirement, and is provided to limit the allowable AFD when $F_Q(X,Y,Z)$ exceeds its limit. This reduces the possibility of operating the core in excess of the $F_Q(X,Y,Z)$ limit when a margin calculation (discussed in item 7 below) indicates negative operational margin exists.

3. There is no change to SR 4.2.2.1.

4. SR 4.2.2.2 addresses obtaining an incore flux map and the requirements based on the results of the measurement. The reference to RAOC operation has been deleted, since RAOC operation is unique to Westinghouse methodology.

5. There is no change to SR 4.2.2.2.a.

6. SR 4.2.2.2.b in the current surveillance has been deleted. The allowances for measurement uncertainty and manufacturing tolerances have been included in the limit [$F_Q^L(X,Y,Z)$] and therefore the measured peak $F_Q^M(X,Y,Z)$ is not increased by these factors.

7. SR 4.2.2.2.c in the current surveillance has been deleted. No simple determination is made of only whether or not the limit has been exceeded. Instead, the amount by which the 4.2.2.2 measured value is above or below the limit is qualified as detailed in item 10, below.

8. SR 4.2.2.2.d (current surveillance) specifies the frequency for measuring the core power distribution. This is done by part b in the new surveillance. Part b.3 has been added to this surveillance, requiring an $F_Q(X,Y,Z)$ measurement when the excore quadrant power tilt ratio is normalized using incore detector measurements. This ensures that the impact of any core tilt on $F_Q(X,Y,Z)$ will be determined and reflected in the margin calculations of part c.

9. SR 4.2.2.2.e has been replaced by SR 4.2.2.2.d in the new surveillance. The intent of these requirements is similar in that projections of the measurements are made to determine at what point peaking would exceed allowable limits if the current trend continues. In the new surveillance, an incore flux map is obtained and a determination is made as to whether the measured $F_Q(X,Y,Z)$ will exceed the allowable peaking at 31 Effective Full Power Days (EFPD) beyond the most recent measurement. If the extrapolated $F_Q(X,Y,Z)$ measurement exceeds the allowable $F_Q(X,Y,Z)$ limit, then either the surveillance interval to the next power distribution map is decreased based on the available margin, or the $F_Q(X,Y,Z)$ measurement is increased by 2% and the margin calculation of 4.2.2.2.c repeated. This surveillance helps ensure that peaking will not exceed allowable limits prior to the next 31 EFPD measurement interval.

10. The new SR 4.2.2.2.c replaces 4.2.2.2.f in the current surveillance. The purpose of part c.1 is to perform margin calculations based on the measured peaks. With the new methodology, the limit ($[F_Q^L(X,Y,Z)]$) to which the measurement is compared is the design peak at steady-state conditions, increased by a factor that represents the maximum amount that the power at the given assembly location and axial elevation can increase above the design value before the measured value may become limiting. Margins to both the LOCA peaking limit (operational margin) and the centerline fuel melt limit (RPS margin) are calculated. The operational margin forms the basis for restricting the AFD limits in part c.2, and the RPS margin forms the basis for reducing the OTΔT trip setpoint in part c.3.

11. SR 4.2.2.2.c.2 (new) replaces SR 4.2.2.2.f.2 in the current surveillance. The reduced AFD limits determined in part c.2 are based on the amount of negative operational margin resulting from the margin calculation of part c.1. The negative and positive AFD limits are reduced 1% for each percent change in margin. The AFD must be controlled to these new limits to reduce $F_Q(X,Y,Z)$, and to ensure that peaking will be limited for continued power operation.

12. SR 4.2.2.2.c.2.b (new) corresponds to SR 4.2.2.2.f.2.b (current surveillance).

13. Part 4.2.2.2.c.3 has been added to the surveillance. This part of the surveillance requires reducing the K_1 value of the OTΔT trip setpoint if the RPS margin is negative. This requirement ensures that centerline fuel melt protection exists when core peaking may be greater than the design values.

14. SR 4.2.2.2.f.2.c, which addresses Base Load operation, has been deleted from the new surveillance. The power distribution methodology of DPC-NE-2011PA does not constrain core operation to a target AFD.

15. SR 4.2.2.2.g has been replaced by SR 4.2.2.2.e in the new surveillance; there are no substantive changes to this surveillance.

16. SR 4.2.2.3 addresses Base Load Operation and has been deleted from the new surveillance.

17. SR 4.2.2.4 addresses surveillance of peaking in Base Load operation and has been deleted from the new surveillance.

18. SR 4.2.2.5 has been replaced by SR 4.2.2.3 in the new surveillance; there are no substantive changes to this surveillance.

<u>Technical Justification</u>

These proposed revisions are the same as those approved for McGuire 2 Cycle 8 (Reference 8).

Proposed Revision to Technical Specification 3/4.2.3

Specification 3/4.2.3 was revised to reflect the power peaking surveillance method described in DPC-NE-2011PA. These revisions are summarized as follows:

1. The statement of the LCO was revised to reflect new nomenclature for the nuclear enthalpy rise hot channel factor [$F^N_{\Delta H}(X,Y)$] and related parameters required by the methodology of DPC-NE-2011PA and used throughout the Reload Report.

2. Those requirements of Actions a and b in the current specification relating to the Reactor Coolant System flow rate have been incorporated in Specification 3.2.5. The Actions have been revised to include the reduction of allowable thermal power when $F^M_{\Delta H}(X,Y)$ exceeds the limit within 2 hours. The factor (RRH), by which the power level is decreased per percent $F_{\Delta H}(X,Y)$ is above the limit, is defined in the COLR. The inverse of this factor is the fractional increase in the MAPs allowed when thermal power is decreased by 1% RTP. When a power level decrease is required because $F_{\Delta H}(X,Y)$ has exceeded its limit, then Action b requires restoration of $F_{\Delta H}(X,Y)$ to within its limit within 6 hours, or a reduction in the high flux trip setpoint. The amount of reduction of the high flux trip setpoint is governed by the same factor (RRH) that determines the thermal power level reduction. This maintains core protection and an operability margin at the reduced power level similar to that at rated thermal power.

3. Action b.3 was replaced by Action d. The portions of the Action requirements related to Reactor Coolant System flow rate have been incorporated in Specification 3.2.5, and do not appear in Action d of the new specification.

4. Action item c has been added and requires a reduction in the OTΔT K_1 trip setpoint by an amount equivalent to TRH for each 1% $F_{\Delta H}(X,Y)$ exceeds its limit within 72 hours of initially being outside the limit. This action ensures that the one protection margin is maintained at the reduced power level for DNB related transients not covered by the reduction in the Power Range Neutron Flux-High Trip Setpoint.

5. There is no change to SR 4.2.3.1.

6. SR 4.2.3.2 formerly covered only surveillance frequency. It has been expanded as detailed below to reflect the power peaking surveillance method described in DPC-NE-2011PA and the format of the revised SR 4.2.2.2. Part a addresses obtaining an incore flux map.

7. SR 4.2.3.2.b (new) replaces the current 4.2.3.2 and addresses the frequency for confirming that $F_{\Delta H}(X,Y)$ is within its limit. In addition to performing the surveillance at least once per 31 EFPD, the revised surveillance requires measurement of the peaking factor whenever the excore quadrant power tilt ratio is normalized using incore detector measurements. This ensures that the impact of any core tilt on $F_{\Delta H}(X,Y)$ will be determined and reflected in the margin calculation. This is comparable to the new SR 4.2.2.2.b in the $F_Q(X,Y,Z)$ specification. The surveillance requiring a surveillance to be performed prior to operation above 75% of RATED THERMAL POWER at the beginning of each fuel cycle

has been replaced by the requirement identical to SR 4.2.2.2.b.2 in the $F_Q(X,Y,Z)$ specification. This surveillance ensures that the plant is at equilibrium conditions prior to a measurement, and also has a provision similar to the requirement it replaced stating that 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.

8. SR 4.2.3.2.c has been added. The purpose of part c.1 is to perform margin calculations based on the measured radial peak. The limit $[F^L_{\Delta H}(X,Y)]^{LCO}$ to which the measurement is compared is based on the allowable design MARP limit, increased by a factor that represents the maximum amount that the power at the given assembly location can increase above the design value before the measured value may become limiting. Part c.2 uses the amount of margin determined by this procedure to form the basis for the amount of power level reduction and the reduction in the high flux and OTΔT K_1 trip setpoints required in the ACTION statements for the specification. This is comparable to the new SR 4.2.2.2.c on $F_Q(X,Y,Z)$.

9. SR 4.2.3.2.d has been added. This surveillance requires projections of the measurements to be made to determine at what point $F_{\Delta H}(X,Y)$ would exceed the allowable limit if the current trend continues. In part d.1 a penalty is applied to $F^M_{\Delta H}(X,Y)$ if the trend indicates that the measured peak would exceed the limiting peak within the 31 EFPD surveillance period, and the margin calculations are repeated. This provides additional margin, or a buffer, to help ensure that the peak will not exceed the limit prior to next 31 EFPD measurement interval. In part d.2, the measurement is obtained and the margin calculations are repeated so that appropriate actions can be taken before zero margin is reached. This surveillance ensures the core is monitored at a frequency that considers conditions when measured peaks are underpredicted. This is comparable to the new SR 4.2.2.2.d on $F_Q(X,Y,Z)$.

10. SR 4.2.3.3, 4.2.3.4, and 4.2.3.5 in the current specification address measurement of Reactor Coolant System flow rate. These requirements have been incorporated in Specification 3.2.5, and have been deleted from the revised requirements for SR 4.2.2.

Technical Justification

These proposed revisions are the same as those approved for McGuire 2 Cycle 8 (Reference 8).

Proposed Revision to Technical Specification 3/4.2.4

This proposed revision is intended to provide Quadrant Power Tilt Ratio limits consistent with Duke Power Company methodology.

Technical Justification

The proposed revisions are the same as the Quadrant Power Tilt Ratio limit changes in the approved submittal for Catawba 1 Cycle 6 (Reference 10).

Proposed Revision to Technical Specification 3/4.2.5

This proposed revision is intended to provide DNB parameter limits consistent with Duke Power Company methodology.

Technical Justification

The proposed revisions are the same as the DNB parameter limit changes in the approved submittals for Catawba 1 Cycle 6 (Reference 10) which revised the DNB parameter limits consistent with Duke Power Company methodology and Catawba 1 Cycle 7 (Reference 9) which corrected a typographical error.

Proposed Revision to Technical Specification Table 3.3-1

This change is to delete the reactor trip on power range neutron flux negative rate from the Reactor Protection System.

Technical Justification

This proposed revision is the same as that approved for Catawba 1 Cycle 7 (Reference 9).

Proposed Revision to Technical Specification Table 3.3-2

The reactor trip on power range neutron flux negative rate is deleted. Neutron detector response time exemption is added to OPΔT trip.

Technical Justification

These proposed revisions are the same as those approved for Catawba 1 Cycle 7 (Reference 9).

Proposed Revision to Technical Specification Table 4.3-1

This change is to delete the reactor trip on power range neutron flux negative rate from the Reactor Protection System.

Technical Justification

This proposed revision is the same as that approved for Catawba 1 Cycle 7 (Reference 9).

Proposed Revision to Technical Specification Table 3.3-4

This proposed revision changes the low steam line pressure setpoint for safety injection and main steam line isolation from 725 psig to 775 psig. The allowable value for this trip function is changed from 694 psig to 744 psig, maintaining the same 31 psig allowance for rack uncertainties, and the lead-lag controller for steam line pressure-low is deleted.

Technical Justification

These proposed revisions are the same as those approved for Catawba 1 Cycle 7 (Reference 9).

Proposed Revision to Technical Specification Table 3.3-5

Two response times are modified in this proposed change, the feedwater isolation response time is changed from 7 seconds to 12 seconds and the steam line isolation time is changed from 7 seconds to 10 seconds.

Technical Justification

These proposed revisions are the same as those approved for Catawba 1 Cycle 7 (Reference 9).

Proposed Revision to Technical Specification 3.3.3.12 & 4.3.3.12.2

It is proposed that the reactor makeup water pump flowrate limit for Mode 5 be changed to 70 gpm in Technical Specification 3.3.3.12(a)(2), 3.3.3.12(b)(2) & 4.3.3.12.2(b)

Technical Justification

Catawba is equipped with a Boron Dilution Mitigation System which serves to detect uncontrolled dilution events in Modes 3 - 6 of plant operation. The BDMS uses two source range detectors to monitor the subcritical multiplication of the reactor core. An alarm setpoint is continually calculated as four times the lowest count rate, including compensation for background and the statistical variation in the count rate. Once the alarm setpoint is exceeded, each train of the BDMS will automatically shut off both reactor makeup water pumps, align the suction of the charging pumps to highly borated water from the Refueling Water Storage Tank, and isolate flow to the charging pumps from the Volume Control Tank. Since these functions are automatically actuated by the BDMS, no operator action is necessary to terminate the dilution event and recover the shutdown margin. In the event one or more trains of the BDMS is inoperable, the reactor makeup water pump flowrate limits ensure that the operator has sufficient time to recognize and terminate a boron dilution event prior to the loss of shutdown margin during each appropriate mode of plant operation. Each cycle, a bounding ratio of initial to critical boron concentration is established from the reload design. This ratio is used to calculate the maximum reactor makeup water pump flowrate which satisfies the operator action time acceptance criteria of the Standard Review Plan. The limits on reactor makeup water pump flowrates when the Boron Dilution Mitigation System (BDMS) is inoperable are verified each cycle to ensure the safety analysis assumptions for these parameters remain valid. When the calculated reactor makeup water flowrate is found to be less than the existing flowrate limits, the flowrate limits must be reduced such that the operator action time acceptance criteria can be met. These cycle-specific parameter limits are verified using the NRC approved methodology provided in the attachment to a Duke Power letter to the U. S. Nuclear Regulatory Commission, "... Supplementary Information Relative to Topical Report BAW-10173; Boron Dilution Analysis", dated May 15, 1991 (Reference [illegible]) and Catawba FSAR (Reference 11) Section 15.4.6.. It is proposed that the reactor makeup water flowrate limit for Mode 5 be reduced to 70 gpm. This new flowrate limit is required to satisfy the operator action time acceptance criteria in the Standard Review Plan.

Proposed Revision to Technical Specification 3.4.2.1 & 3.4.2.2

This modification changes the tolerances on the pressurizer safety valve lift setpoint from ±1% to +3%,-2% in all modes of operation.

Technical Justification

These proposed revisions are the same as those approved for Catawba 1 Cycle 7 (Reference 9).

Proposed Revision to Technical Specification Table 3.6-2a & 3.6-2b

This change clarifies the required maximum stroke time of the steam generator main feedwater to auxiliary feedwater nozzle isolation valves, auxiliary nozzle temper valves, steam generator feedwater containment isolation valves, steam generator feedwater purge valves, main steam isolation valves, and main steam isolation bypass control valves. The numerical value of the stroke time of these valves is changed to NA.

Technical Justification

The justification for the change in valve stroke time as it relates to system thermal-hydraulic response during a steam line break event was presented for a change to Technical Specification Table 3.3-5 in the Catawba Unit 1 Cycle 7 reload submittal (Reference 9). Although these valves are included in Tables 3.6-2a and 3.6-2b, the list of containment isolation valves for Unit 1 and Unit 2, these valves do not receive a containment isolation signal. As shown in Catawba FSAR Figure 7-2, Part 8 of 16, a containment pressure high signal, low pressurizer pressure signal, low steamline pressure signal or a safety injection signal will actuate feedwater isolation in addition to and separate from a phase "a" isolation. Also, a containment high-high signal, low steamline pressure, or high steam pressure rate signal will actuate a steamline isloation in addition to and separate from a phase "b" isolation. The valves in the proposed change are "actuated by signal other than S, T, or P signal (main steam isolation, feedwater isolation , low RN pit level,...)" according to note 8 of Catawba FSAR Table 3-104. These valves perform a containment isolation function only to the extent that credit for their operation might be taken in the dose analysis. Since these valves receive no containment isolation signal, and credit for the operation of these valves is not taken in the dose analysis, a maximum stroke time is not applicable for these valves.

Proposed Revision to Technical Specification 4.7.1.4

The permissible stroke time for the main steam isolation valves is changed from 5 to 8 seconds.

Technical Justification

This proposed revision is the same as that approved for Catawba 1 Cycle 7 (Reference 9).

Proposed Revision to Technical Specification 6.9.1.9

Add NRC approved Topical DPC-NE-1004A, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P" to list of analytical methods used to determine the core operating limits.

Technical Justification

This change is administrative in nature since it updates the reference list with a newly approved topical describing methodology used to determine core operating limits.

References

1. BAW-10159P-A, BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies, Babcock & Wilcox, July 1990.

2. DPC-NE-2011P-A, Duke Power Company, Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors, March, 1990.

3. BAW-10174-A, Mark-BW Reload LOCA Analysis for the Catawba and McGuire Units, Babcock & Wilcox, May 1991.

4. DPC-NE-3001P, Duke Power Company, Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, Revision 1, November 1991.

5. DPC-NE-2004P-A, Duke Power Company, McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01, December 1991.

6. WCAP-10988, Cobra-NC, Analysis for a Main Steamline Break in the Catawba Unit 1 Ice Condenser Containment, Westinghouse Nuclear Energy Systems, November 1985.

7. McGuire Nuclear Station Unit 1, Docket Number 50-369, Cycle 8 Reload Submittal, Duke Power Company, June 26,1991.

8. McGuire Nuclear Station Unit 2, Docket Number 50-370, Cycle 8 Reload Submittal, Duke Power Company, December 18,1991.

9. Catawba Nuclear Station Unit 1, Docket Numbers 50-413 and 50-414, Cycle 7 Reload Submittal, Duke Power Company, April, 13, 1992.

10. Catawba Nuclear Station Unit 1, Docket Numbers 50-413 and 50-414, Cycle 6 Reload Submittal, Duke Power Company, January 9, 1991.

11. Catawba Nuclear Station, Final Safety Analysis Report, Docket Nos. 50-4J3/414.

12. Duke letter to U. S Nuclear Regulatory Commission, McGuire Nuclear Station Docket Numbers 50-369 and -370 Catawba Nuclear Station Docket Numbers 50-413 and -414 Supplementary Information Relative to Topical Report BAW-10173; Boron Dilution Analysis, Duke Power Company, May 15,1991.

13. McGuire Nuclear Station, Final Safety Analysis Report, Docket Nos. 50-369/370.

Attachment 3

No Significant Hazards Evaluation

10 CFR 50.92 states that a proposed amendment involves no significant hazards consideration if operation in accordance with the amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in a margin of safety.

<u>CHANGES WHICH ARE THE SAME AS THOSE MADE FOR C1C7</u>

The changes to the Safety Limit and Power Distribution Technical Specifications presented in Section 8 of the Reload Report represent the application of previously approved methodology to Catawba Unit 2. The changes to remove the power range neutron flux negative rate reactor trip, increase the low steam line pressure setpoint, increase feedwater isolation response time, increase steam line isolation response time, increase pressurizer safety valve lift setpoint tolerance, remove steam line pressure dynamic compensation, increase pressurizer safety valve lift setpoint tolerance, and increase main steam line isolation valve stroke time reflect the use of Duke analysis, and have already been approved for Catawba Unit 1. The changes described above include the deletion of references to specific units on individual Technical Specification pages, and delete pages which were previously for Unit 2 only. The implementation of unit specific references became necessary due to the transition from Westinghouse to B&W supplied fuel and for the Cycle 7 Reload due to the transition to Duke analysis methodology. The analysis which made the changes necessary in the Unit 1 reload submittal is generic, and as described in the technical justification, is equally applicable to both McGuire and Catawba units. Therefore, there is no new significant hazards consideration which will be raised by this amendment. This determination is in keeping with staff guidance which was published in the Federal Register (48FR14864) to assist in determining whether or not proposed amendments are likely to raise a significant hazards consideration. This guidance cites as an example of an amendment not likely to involve a significant hazards consideration "a purely administrative change to technical specifications: for example, a change to achieve consistency..."

Since these changes are considered administrative, no further analysis is required.

<u>CHANGES TO TS 3/4.6.3</u>

The proposed changes to the valve stroke times in Table 3.6-2a and 3.6.2b will not significantly increase the probability or consequences of any previously evaluated accident. The effects of the delays in isolation times on the various transients affected have been analyzed and found to be acceptable. Since these valves do not receive a containment isolation signal, and no credit is taken for operation of these valves in the dose analysis for a containment isolation function, a maximum stroke time does not apply for containment isolation.

The proposed changes will not significantly increase the possibility of a new accident not previously evaluated. Feedwater and main steam isolation are responses to ongoing transients, rather than initiators or precursors of transients. No equipment or component reconfiguration will occur as a result of this change.

The proposed changes will not significantly decrease any margin of safety. The isolation times which are applicable to these valves are specified in Table 3.3-5, Engineered Safety Features Response Times. The effects of the isolation of these valves was evaluated based on their ESF function, not a containment isolation function, and determined to be acceptable, therefore there is no significant decrease in the margin of safety.

CHANGE TO TS 3.3.3.12.a.2

TS 3.3.3.12.a.2 is changed to reduce the allowable Reactor Makeup Water Pump flow in Mode 5 from 75 gpm to 70 gpm. In the event that the Boron Dilution Mitigation System (BDMS) is inoperable the Reactor Makeup Water Pump flowrates are limited to ensure that operator action times required to terminate a dilution event can be met. The limits on reactor makeup water pump flowrates when the BDMS is inoperable are verified each cycle to ensure that the safety analysis assumptions for these parameters remain valid. When the calculated Reactor Makeup Water Pump flowrate is found to be less than the existing flowrate limits, the flowrate limit must be reduced so that the operator action time acceptance criteria of Standard Review Plan 15.4.6 can be met.

Reducing the allowable Reactor Makeup Water Pump flow in Mode 5 does not involve a significant increase in the probability or consequences of an accident previously evaluated. The current TS flowrate does not allow enough time for the operator to terminate an uncontrolled dilution event when required operator response times are assumed. The lower flowrate allows needed operator response times and is therefore more conservative.

Reducing the allowable Reactor Makeup Water Pump flow in Mode 5 does not change the way that any plant equipment is operated or maintained, therefore it does not create the possibility of a new or different accident.

Reducing the Allowable Reactor Makeup Water Pump Flow in Mode 5 will not involve a significant reduction in the margin of safety. This flowrate is more conservative, and ensures that safety analysis assumptions regarding operator actions times in response to the termination of an uncontrolled dilution event can be met.

Changes to TS 6.9.1.9

The proposed change to TS 6.9.1.9 adds approved topical DPC-NE-1004A to the list of analytical methods used to determine core operating limits. This change is administrative, adding a topical report which has been approved for use on Catawba to the list of analytical methods used to determine core operating limits. Since this change is administrative it has been determined that no significant hazards are involved.

The proposed Technical Specification change has been reviewed against the criteria of 10 CFR 51.22 for environmental considerations. As shown above, the proposed change does not involve any significant hazards consideration, nor increase the types and amounts of effluents that may be released offsite, nor increase the individual or cumulative occupational radiation exposures. Based on this, the proposed Technical Specification change meets the criteria given in 10 CFR 51.22(c)(9) for categorical exclusion from the requirement for an Environmental Impact Statement.