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RELOAD REPORT

Catawba Unit 1, Cycle 6

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the sixth cycle of Catawba Nuclear Station, Unit 1 at the rated core power level of 3411 MWt. Included are the required analyses as outlined in the USNRC document "Guidance for Proposed License Amendments Relating to Refueling," July 1975.

Cycle 6 for Catawba Unit 1 will be the first cycle for which the reload fuel is supplied by B&W Fuel Company (BWFC) and will therefore be the reference cycle for BWFC fuel in the Catawba units. The incoming batch 8 fuel assemblies are designated as Mark-BW. To support implementation of Mark-BW fuel in the Catawba and McGuire nuclear plants, BWFC has developed new methods and models to analyze the plants during normal and off-normal operation. These methods and models are documented in topical reports and have been reviewed by the NRC. Most of the topical reports have already been approved. Approval of the final four topical reports is scheduled for completion by December 14, 1990.

Section 2 of this report is the operating history for fuel in Catawba Unit 1. Section 3 is a general description of the reactor core, and the fuel system design is provided in Section 4. Reactor and system parameters and conditions are summarized in Sections 5, 6, and 7. Changes to the Technical Specifications, Core Operating Limits Report (COLR), and Final Safety Analyis Report are provided in Section 8. The scope of Physics Startup Testing for Catawba Unit 1, Cycle 6 is provided in Section 9.

All of the accidents analyzed in the FSAR[1] have been reviewed for Cycle 6 operation. In those cases where Cycle 6 characteristics were conservative compared to those analyzed for previous cycles, new analyses were not performed. Several bounding transients were analyzed in detail to demonstrate the capability of BWFC calculational techniques. The results of these analyses were reported in BAW-10173P.[2]

On May 17, 1990 the NRC issued Amendment Number 74 and Amendment Number 68 to the Catawba Nuclear Station Facility Operating License. These amendments allow the removal of cycle-specific core parameter limits from Technical Specifications and require that these limits be included in a Core Operating Limits Report (COLR). The Core Operating Limits Report is submitted to the NRC upon issuance and does not require approval prior to implementation. Changes to the operating limits are made via the Core Operating Limits Report.

The Technical Specifications have been reviewed, and the modifications for Cycle 6 are justified in this report. Based on the analyses performed, which take into account the postulated effects of fuel densification and the Final Acceptance Criteria for emergency core cooling (ECC), it has been concluded that Catawba Unit 1 Cycle 6 can be safely operated at a core power level of 3411 MWt.

2. OPERATING HISTORY

The current operating cycle for Catawba Unit 1 is Cycle 5 which achieved criticality on April 22, 1990 and reached 100% full power on April 29, 1990. Cycle 5 is scheduled to shut down in March 1991 after 300 EFPD. This cycle and all previous cycles have operated with fuel assemblies of the Westinghouse design.

Cycle 6 will be the new reference cycle and will be the first fuel cycle containing BWFC Mark-BW fuel assemblies (FAs). It is scheduled to start up in June 1991 at a rated power level of 3411 MWt and has a design cycle length of 350 EFPD. No operating anomalies have occurred during previous cycle operations that would adversely affect fuel performance in Cycle 6.

3. GENERAL DESCRIPTION

The Catawba Unit 1 reactor core is described in detail in chapter 4 of the FSAR[1]. The core consists of 193 fuel assemblies, each of which is a 17-by-17 array containing 264 fuel rods, 24 guide tubes, and one incore instrument guide tube. The 121 burned FAs are of the Westinghouse Optimized Fuel Assembly (OFA) design, and the 72 fresh FAs are of the Mark-BW design[3]; all the fuel rods have Zircaloy-4 cladding. The fuel rod outside diameters are 0.360 and 0.374 inch, and the wall thicknesses are 0.0225 and 0.024 inch for the OFA and Mark-BW designs, respectively. The Mark-BW fuel consists of dished-end, cylindrical pellets of uranium dioxide (see Table 4-1 for data). The average nominal fuel loadings are 423.119, 424.898, 423.119, and 456.300 kg of uranium per fuel assembly in batches 3D, 6B, 7, and 8 respectively. Figure 3-1 is the core loading diagram for Cycle 6 of Catawba Unit 1. The initial enrichments of batches 3D, 6B, and 7 were 3.10, 3.279, and 3.40 wt % ^{235}U, respectively. The design enrichment of fresh batch 8 is 3.55 wt % ^{235}U.

The fifty-two batch 6B and sixty-eight batch 7 assemblies will be shuffled to new locations. One batch 3D FA discharged at the end of Cycle 2 will be re-inserted as the center assembly. The seventy-two fresh batch 8 assemblies will be loaded into the core in a symmetric checkerboard pattern. Figure 3-2 is an eighth-core map showing the burnup and initial enrichment of each assembly at the beginning of Cycle 6.

Cycle 6 will be operated in a feed-and-bleed mode. Core reactivity is controlled by 53 rod cluster control assemblies (RCCAs), 52 BPRAs, and soluble boron shim. The Cycle 6 locations of the 53 rod cluster control assemblies with their respective designations are indicated in Figure 3-3. The Cycle 6 locations and number of pins of 3.0 wt % B_4C per BPRA cluster are shown in Figure 3-4.

Figure 3-1. Core Loading Diagram for Catawba Unit 1 Cycle 6

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					7 J14	8 F	7 E14	8 F	7 L14	8 F	7 G14				
2			6B N6	6B C14	8 F	7 M13	8 F	7 H5	8 F	7 D13	8 F	6B N14	6B C6		
3		6B K3	7 K5	8 F	6B B4	8 F	6B A7	7 H15	6B R7	8 F	6B P4	8 F	7 F5	6B F3	
4		6B B13	8 F	7 E12	8 F	6B F2	8 F	7 D7	8 F	6B K2	8 F	7 L12	8 F	6B P13	
5	7 B7	8 F	6B M14	8 F	[illegible] L6	7 K15	6B F7	8 F	6B K7	7 F15	7 E6	8 F	6B D14	8 F	7 P7
6	8 F	7 C4	8 F	6B P10	7 A6	6B C13	8 F	7 H9	8 F	6B N13	7 R6	6B B10	8 F	7 N4	8 F
7	7 B11	8 F	6B J15	8 F	6B [illegible]	8 F	7 J4	7 M11	7 G4	8 F	6B G10	8 F	6B C15	8 F	7 P11
8	8 F	7 L8	7 A8	7 M7	8 F	7 G8	7 D11	3D J9 CY2	7 M5	7 J8	8 F	7 D9	7 R8	7 E8	8 F
9	7 B5	8 F	6B J1	8 F	6B J6	8 F	7 J12	7 D5	7 G12	8 F	6B G6	8 F	6B G1	8 F	7 P5
10	8 F	7 C12	8 F	6B P6	7 A10	6B C3	8 F	7 H7	8 F	6B N3	7 R10	6B B6	8 F	7 N12	8 F
11	7 B9	8 F	6B M2	8 F	7 L10	7 K1	6B F9	8 F	6B K9	7 F1	7 E10	8 F	6B D2	8 F	7 P9
12		6B B3	8 F	7 E4	8 F	6B F14	8 F	7 M9	8 F	6B K14	8 F	7 L4	8 F	6B P3	
13		6B K13	7 K11	8 F	6B B12	8 F	6B A9	7 H1	6B R9	8 F	6B P12	8 F	7 F11	6B F13	
14			6B N10	6B C2	8 F	7 M3	8 F	7 H11	8 F	7 D3	8 F	6B N2	6B C10		
15					7 J2	8 F	7 E2	8 F	7 L2	8 F	7 G2				

xxx
yyy
zzz

Key:
xxx - Batch ID
yyy - Previous cycle location
zzz - Previous cycle (if reinsert)

Figure 3-2. Enrichment and BOC Burnup Distribution for Catawba Unit 1 Cycle 6

	H	G	F	E	D	C	B	A
8	3.10 27,098	3.40 16,567	3.40 16,660	3.55 0	3.40 16,368	3.40 10,434	3.40 16,700	3.55 0
9		3.40 16,362	3.55 0	3.279 27,965	3.55 0	3.279 24,997	3.55 0	3.40 13,453
10			3.279 26,318	3.40 9,383	3.279 29,109	3.55 0	3.40 14,820	3.55 0
11				3.40 16,639	3.55 0	3.279 23,089	3.55 0	3.40 14,895
12					3.40 16,546	3.55 0	3.279 20,688	
13						3.40 16,626	3.279 24,822	
14								
15								

X.XX	Enrichment, Initial
XX,XXX	Burnup, (MWd/mtU), BOC

Figure 3-3. Rod Cluster Control Assembly Locations and Bank Designations for Catawba Unit 1 Cycle 6

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2				SA		CB		CC		CB		SA			
3					SD		SB		SB		SC				
4		SA		CD				SE				CD		SA	
5			SC										SD		
6		CB				CC		CA		CC				CB	
7			SB										SB		
8		CC		SE		CA		CD		CA		SE		CC	
9			SB										SB		
10		CB				CC		CA		CC				CB	
11			SD										SC		
12		SA		CD				SE				CD		SA	
13					SC		SB		SB		SD				
14				SA		CB		CC		CB		SA			
15															

Key:

xx - Rod bank identifier

Figure 3-4. Burnable Poison Pin Distribution for Catawba Unit 1 Cycle 6

	H	G	F	E	D	C	B	A
8				12				
9			12		12		8	
10		12				12		
11	12				12			
12		12		12		4		
13			12		4			
14		8						
15								

Pins per BPRA	No. of BPRAs	No. of pins
4	8	32
8	8	64
12	[illegible]	432
TOTAL	52	528

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The Catawba 1 Cycle 6 batch 8 feed comprises 72 Mark-BW fuel assemblies with an enrichment of 3.55 wt% ^{235}U. A total of 52 Mark-BW BPRA's are used with either 4, 8 or 12 BP pins each. The remainder of the fuel assemblies in Cycle 6 are Westinghouse Optimized Fuel Assemblies. The Mark-BW fuel assembly is a 17x17, standard lattice, Zircaloy spacer grid fuel assembly designed for use in Westinghouse designed reactors. The fuel assembly incorporates many standard BWFC design features while maintaining compatibility with the Westinghouse reactor internals and resident fuel assemblies. The nozzles, nozzle attachment and end spacer grids are based on proven Nuclear Fuel Industries (NFI) designs currently in operation in Westinghouse-designed reactors in Japan. The guide thimble top section, dashpot diameters, instrument sheath diameter, and the fuel rod outside diameter are the same as the standard 17x17 Westinghouse design. The fuel rod design has been developed based on standard BWFC methods applied to the Westinghouse standard outside cladding diameter. The unique features of the Mark-BW fuel assembly design include the Zircaloy intermediate spacer grid, the spacer grid restraint system, and the use of Zircaloy spacer grids with the standard lattice design.

The fuel assembly, shown in Figure 4-1, consists of 264 fuel rods, 24 guide thimbles, and one instrument sheath in a 17x17 square array. The guide thimbles provide guidance for RCCA insertion and are attached to nozzles at the top and bottom of the fuel assembly and to the bottom end spacer grid to form the structural skeleton. A reduced diameter section at the bottom of the guide thimbles decelerates the RCCA during trips. The instrument sheath occupies the center lattice position and provides guidance and protection for the incore instrumentation assemblies. The top nozzle assembly contains the fuel assembly holddown springs and is attached to the guide thimbles by removable nuts and locking cups. The bottom nozzle is attached to the guide thimbles by bolts which are mechanically captured by tack welding. The bottom nozzle utilizes a unique flow hole pattern consisting of small round holes and a cloverleaf shaped hole to prevent passage of debris which could lead to fuel rod failure due to fretting. The fuel rod and guide thimble spacing are maintained along the length of the assembly by six Zircaloy intermediate spacer grids.

4.2 Fuel Rod Design

Analyses were performed on the Mark-BW fuel rod design to assure that its mechanical performance in-reactor would be adequate; the methods are described in Reference 3. The areas that were analyzed are:

A. Creep Collapse

B. Cladding Stress

C. Cladding Strain

D. Cladding Fatigue

4.2.1 Fuel Rod Cladding Collapse

The fuel rods were analyzed for creep collapse using methods outlined in Reference 3 and the creep collapse code CROV[4]. Using nuclear design inputs, a power history was determined which enveloped all past fuel rod operating conditions for the Catawba Plant. This power history with appropriate uncertainty factors was input into the computer code TACO2[5] which determined the temperature, the pressure, and the fast neutron flux level history of the Mark-BW fuel rods. This history was input to CROV using conservative cladding dimensions. From the output of CROV the creep collapse point of the Mark-BW fuel rods was determined to be greater than 60,000 MWd/mtU. This burnup exceeds the maximum burnup and exposure the Mark-BW fuel rods are expected to experience in Catawba 1, Cycle 6.

4.2.2 Fuel Rod Cladding Stress

The fuel rod cladding was analyzed for the stresses induced during Condition I and II operation. The ASME pressure vessel stress intensity limits were used as guidelines. Conservative values were used for cladding thickness, oxide layer buildup, external pressure, internal fuel rod pressure, differential temperature and unirradiated cladding yield strength. The analysis results show that the maximum cladding stress intensities are within limits under all loading conditions.

Stress Analysis Summary

Condition	Limit	S	Margin,%
Pm	Sm (30,000 psi)	22,926 psi	30.9
Pm+Pb	1.5Sm (45,000 psi)	26,272 psi	71.3
Pm+Pb+Pl	1.5Sm (45,000 psi)	27,888 psi	61.4
Pm+Pb+Pl+Q	3.0Sm (90,000 psi)	65,137 psi	38.2

Where:

Pm = general primary membrane stress intensity,
Pb = primary bending stress intensity,
Pl = local primary membrane stress intensity,
Q = secondary stress,
Sm = allowable membrane stress = 2/3 Sy or 1/3 Su (whichever is less),
Sy = yield stress, and
Su = ultimate stress.

4.2.3 Fuel Rod Cladding Strain

The fuel rod was analyzed to determine the maximum allowable local power change that the fuel rod could experience before the very conservative transient strain limit of 1% is exceeded. The transient strain limit is based on diametral cladding strain resulting from a local power transient. The maximum allowable local power change determined from the analysis was compared to the maximum calculated local power change induced by a worst case core maneuvering scenario. This comparison determined that margin exists to the 1% strain limit.

4.2.4 Fuel Rod Cladding Fatigue

The fuel rod was analyzed for the total fatigue usage factor using the ASME pressure vessel code as a guideline. A maximum fatigue usage factor of 0.9 is allowed. A fuel rod life of 8 years was assumed. All possible Condition I & II events expected and one Condition III event were analyzed to determine the total fatigue usage factor experienced by the fuel rod cladding. Conservative inputs in terms of cladding thickness, oxide layer buildup, external pressure, internal fuel rod pressure and differential temperature across the cladding were assumed. The results of the fatigue analysis show a maximum fatigue usage factor of 0.35.

4.3 Thermal Design

The thermal performance of the fresh batch 8 Mark-BW assemblies was evaluated with the TACO3 code as described in BAW-10162P-A[6]. Nominal undensified input parameters used in the analysis are presented in Table 4-1. Densification effects were accounted for in TACO3.

The results of the thermal design evaluation of the Mark-BW fuel are summarized in Table 4-1. Cycle 6 core protection limits for the Mark-BW fuel are based on a linear heat rate (LHR) to centerline fuel melt limit of 21.86 kW/ft as determined by the TACO3 code.

The maximum fuel assembly burnup at EOC 6 is predicted to be less than 18000 MWd/mtU for the Mark-BW fuel (batch 8). The fuel rod internal pressures have been evaluated with TACO3 for the highest burnup rods and are predicted to be less than the nominal reactor coolant pressure of 2280 psia.

4.4 Material Compatibility

The compatibility of all possible fuel-cladding-coolant-assembly interactions for batch 8 feed fuel assemblies is identical to that of present fuel assemblies.

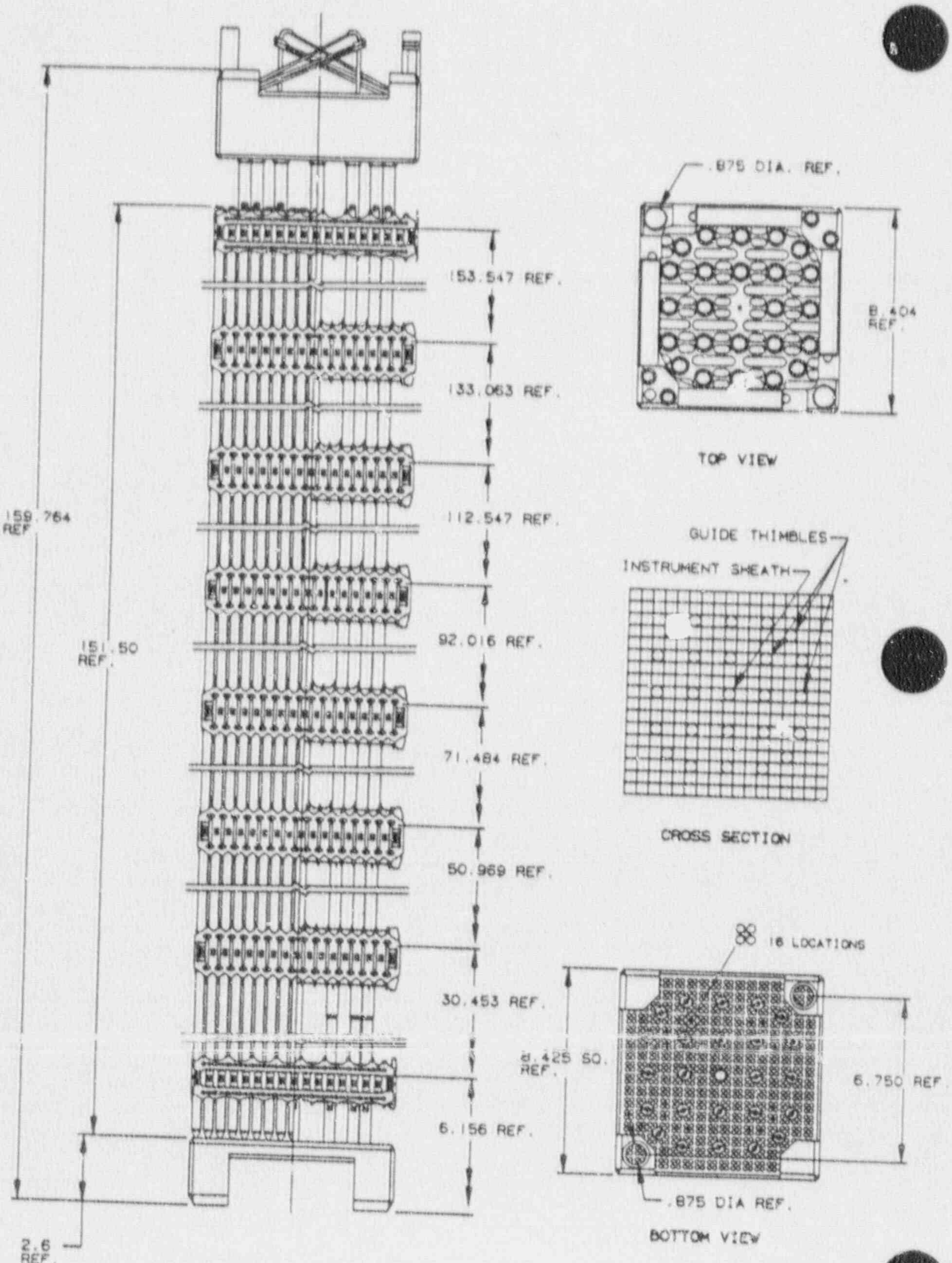
4.5 Operating Experience

Catawba Unit 1 Cycle 6 is the first complete reload batch of Mark-BW 17x17 fuel assemblies. BWFC experience with fuel assembly irradiation extends to 58.3 GWd/mtU for the Mark-B 15x15 product line with over 5,000 assemblies irradiated. Experience with the Mark-BW 17x17 fuel assemblies for Westinghouse designed reactors consists of two sets of lead assemblies in two reactors. This experience started with the irradiation of four lead assemblies in McGuire Unit 1 Cycle 5. The McGuire lead assemblies [illegible] currently in the third cycle of irradiation with an anticipated assembly burnup of 42.8 GWd/mtU by the end of Cycle 7. The McGuire lead assemblies were examined poolside after Cycle 5 and Cycle 6. The last post irradiation examination on the McGuire lead assemblies was done after Cycle 6 at a burnup of 27.6 GWd/mtU. Fuel assembly bow, twist, growth and holddown spring set were within nominal bounds. Four other lead assemblies are undergoing their first cycle of irradiation in Trojan Cycle 13.

Table 4-1. Fuel Design Parameters and Dimensions - Mark BW

Nominal Fuel Rod O.D. (in)	.374
Nominal Fuel Rod I.D. (in)	.326
Nominal Active Fuel Length (in)	144.0
Nominal Fuel Pellet O.D. (in)	.3195
Fuel Pellet Initial Density (%)	96
Initial Fuel Enrichment (wt% ^{235}U)	3.55
Average Burnup BOC, (MWd/mtU)	0
Estimated Residence Time EOC, EFPH	6000
Cladding Collapse Time, EFPH	>30,194
Cladding Collapse Burnup, MWd/mtU	>60,000
Nominal Linear Heat Rate, (kW/ft)	5.43
Average Fuel Temperature at Nominal LHR BOL, (^oF)	1230
Minimum LHR to Melt, (kW/ft)	21.86

Figure 4-1. MARK-BW 17 FUEL ASSEMBLY



5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 provides the core physics parameters for Cycles 5 and 6. The Cycle 6 values were generated using the NOODLE[7] code (see Section 5.2) and are valid for the design cycle length (350 EFPD) plus 10 EFPD. Figure 5-1 illustrates a representative relative power distribution for the beginning of Cycle 6 at full power. This case was calculated by the PDQ07[8] code (see Section 5.2) as part of the design depletion and contained equilibrium xenon and nominal rod positions.

During verification of the control rod insertion limits specified in the COLR, calculated ejected rod worths and their adherence to acceptance criteria were considered. The adequacy of the shutdown margin with Cycle 6 stuck rod worths is demonstrated in Table 5-2. The shutdown margin calculations include allowance for potential flux redistribution and a 10% uncertainty on net rod worth. Flux redistribution was accounted for separately since the shutdown analysis was performed using a two-dimensional model. The shutdown calculation at the end of Cycle 6 was analyzed at 360 EFPD.

5.2. Changes in Nuclear Design

The core design changes for Cycle 6 include the use of Mk-BW fuel assemblies and BPRAs with variable numbers of poison rods as described in Section 3. Additionally, the Cycle 6 design lifetime of 350 EFPD represents an increase compared to previous cycles.

The Cycle 6 physics parameters appearing in this report were calculated with the PDQ07, FLAME3[9], and NOODLE codes. These codes were run in either two or three dimensions depending on the amount of modeling detail required and the characteristics of the individual codes. The PDQ07 calculations were performed in two dimensions; the FLAME3 calculations were performed in three dimensions, and the NOODLE calculations were carried out in both two and three dimensions. The Reactor Protection System (RPS) limits (Technical Specification changes) verified by analyses for this fuel cycle are presented in Section 8. Operational limits for the core are provided in the COLR; revisions to the COLR for Cycle 6 are presented in Section 8.

Table 5-1. Physics Parameters[a], Catawba 1 Cycles 5 and 6

	Cycle 5	Cycle 6
Design cycle length, EFPD	300	350
Design cycle burnup, MWd/mtU	12,520	14,145
Design average core burnup -- EOC, MWd/mtU	25,729	26,216
Design initial core loading, mtU	81.7	84.1
Critical boron -- BOC, ppmb, no Xe[b]		
HZP, all rods out	1,483	1,699
HZP, bank D inserted	1,412	1,630
HFP, all rods out	1,358	1,543
Critical boron -- EOC, ppmb, eq Xe		
HZP	263	290
HFP	15	16
Control rod worths -- BOC, HFP, eq Xe, pcm		
Bank D	647	639
Bank C	1282	1117
Control rod worths -- EOC[c], HFP, eq Xe, pcm		
Bank D	576	703
Bank C	1161	1187
Max ejected rod worth[d] -- HZP, pcm		
BOC (D12)	[e]	355
EOC[c] (D12)	[e]	473
Max stuck rod worth -- HZP, pcm		
BOC (F8)	1065	867
EOC[c] (F8)	1173	912
Power deficit -- HZP to HFP, pcm		
BOC	-1716	-1564
EOC[c]	-3058	-2752
Doppler coeff -- HFP, pcm/°F		
BOC, no Xe	-1.42	-1.57
EOC[c], eq Xe	-1.71	-1.84
Moderator coeff -- HFP, pcm/°F		
BOC, 1670 ppmb, no Xe	-1.53	-1.70
EOC[c], 0 ppmb, eq Xe	-29.38	-32.27

Table 5-1. Physics Parameters[a], Catawba 1 Cycles 5 and 6 (Continued)

	Cycle 5	Cycle 6
Boron worth -- HFP, pcm/ppmb		
BOC	-9.06	-7.85
EOC	-10.25	-9.40
Xenon worth -- HFP, pcm		
BOC (4 EFPD)	2728	2665
EOC[c] (equilibrium)	2991	2837
Effective delayed neutron fraction -- HFP		
BOC	0.00609	0.00626
EOC[c]	[e]	0.00525

(a) Cycle 6 data are for the conditions stated in this report; the Cycle 5 values given were provided by Duke Power Company from their analysis.

(b) HZP denotes hot zero power (557F T_{avg}); HFP denotes hot full power (593F T_{avg}).

(c) EOC physics parameters calculated at design EOC plus 10 EFPD.

(d) Ejected rod worth for banks D, C and B inserted.

(e) These values were not generated by Duke Power Company for Cycle 5.

Table 5-2. Shutdown Margin Calculation for Catawba 1 Cycle 6

Available Rod Worth	BOC, %Δρ	EOC[a], %Δρ
1. Total rod worth, HZP	6.57	7.05
2. Maximum stuck rod worth, HZP	-0.87	-0.91
3. Net Worth	5.70	6.14
4. Less 10% uncertainty	-0.57	-0.61
5. Total available worth	5.13	5.53
Required Rod Worth		
6. Power defect, HFP to HZP	1.35	2.06
7. Max allowable inserted rod worth[b]	0.43	0.65
8. Flux redistribution	0.21	0.70
9. Total required worth	1.99	3.41
10. Shutdown Margin (total avail. worth minus total required worth)	3.13	2.12

NOTE: Required shutdown margin is 1.30%Δρ.

[a] EOC physics parameters calculated at 360 EFPD, i.e., design EOC plus 10 EFPD.

[b] Includes allowance for consideration of RCCA positions as fully withdrawn at 222 steps withdrawn.

Figure 5-1. BOC (4 EFPD), Cycle 6 Two-Dimensional Relative Power Distribution - HFP, Equilibrium Xenon

	H	G	F	E	D	C	B	A
8	0.87	1.12	1.19	1.30	1.20	1.23	1.08	0.88
9		1.17	1.29	0.95	1.25	0.99	1.23	0.71
10			0.99	1.21	0.92	1.22	1.06	0.80
11				1.17	1.27	0.97	1.12	0.46
12					1.15	1.17	0.55	
13						0.69	0.26	
14								
15								

X.XX	Relative power density

6. THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design evaluation supporting Cycle 6 operation was performed with the statistical core design (SCD) analysis method, which incorporates the BWCMV CHF correlation. The SCD method and BWCMV have been demonstrated to be generically applicable in References 10 and 11, respectively. Cycle 6 is the first transition cycle to the Mark-BW design at Catawba 1 and the first cycle to utilize BWFC's SCD methods. Core safety limits for Cycle 6 are based on a full core Mark-BW analysis with a 1.55 design $F^N_{\Delta H}$. In the mixed core, those limits have been confirmed to be applicable to the Westinghouse OFA with a design $F^N_{\Delta H}$ of 1.49. Table 6-1 provides a summary of the thermal-hydraulic design parameters used to evaluate Cycle 6.

The SCD method that was used for this reload evaluation treats uncertainties in design inputs statistically. By doing this, a statistical DNBR design limit is determined that is greater than the BWCMV CHF correlation limit documented in Reference 11. To provide additional design flexibility, a thermal design limit is established that incorporates thermal margin. For the Catawba 1 core the statistical design limit (SDL) has been calculated as 1.345 (BWCMV) based on the plant specific uncertainties listed in Table 6-2. Other generic and fuel dependent uncertainties are the same as those presented in BAW-10170P-A[10]. For the Catawba 1 Cycle 6 analyses the thermal design limit (TDL) is 1.50 BWCMV. The thermal margin based on these values is as follows:

$$\text{Thermal Margin (\%)} = \frac{1.50 - 1.345}{1.50} \times 100 = 10.3\%$$

Table 6-3 outlines the penalties and offsets that must be assessed against the thermal margin included in the TDL.

Table 6-1. Nominal Thermal-Hydraulic Design Conditions, Cycle 6

Design Power Level, MWt	3411
Core Exit Pressure, psia	2280
Nominal Average Temperature, °F	590.8
Reactor Coolant System Flow, gpm	385000
Core Bypass Flow, %	7.5
DNBR Modeling	SCD
Reference Design Radial-Local Power Peaking Factor	Mark-BW 1.55 OFA 1.49
Reference Design Axial Flux Shape	1.55 Cosine
Active Fuel Length, in	144.0
CHF Correlation	BWCMV
Statistical Design Limit (SDL)	1.345
Thermal Design Limit (TDL)	1.50

Table 6-2. Statistical Core Design Application Summary

Measurement Uncertainties

Variable	Name	Uncertainty	Distribution
Q	Core Power	2%	Normal
W	Core Flow	2.2%	Uniform
P	Core Pressure	30 psi	Uniform
T	Core Inlet Temperature	4°F	Uniform
R	Measured $F_{\Delta H}^{N}$	5%	Normal

BWFC Analysis Uncertainties

Variable	Name	Uncertainty	Distribution
W	Core Bypass Flow	1.5%	Uniform
R	Hot Channel Factor	3%[a]	Normal

[a] Also applies to the Westinghouse OFA

Table 6-3. Statistical Core Design Application Summary

Penalties & Offsets to be Assessed Against The Thermal Margin Included in the Thermal Design Limit

Statistical Design Limit (SDL)	1.345
Thermal Design Limit (TDL)	1.50
Percent Margin Available	10.3

Penalty/Offset	Value	
	Mark-BW	OFA
Transition Core	0%	0% [a]
Rod Bow	0%	2.7%
Flow Anomaly [b]	3.5%	3.5%
Instrumentation/Hardware	4.2%	2.8%
Total	7.7%	9.0%
Available DNBR Margin	2.6%	1.3%

[a] OFA evaluations are based on a design $F^N_{\Delta H}$ of 1.49.

[b] The flow anomaly penalty is applied to compensate for an anomalous flow condition that has been detected in the Catawba units. The flow anomaly results from a vortex formed in the lower reactor vessel internals due to the flow distribution and internals configuration.

7.0 ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

In order to determine the effects of this reload and to ensure that the thermal performance during hypothetical incidents is not degraded, each FSAR accident analysis has been evaluated.

The safety analysis evaluation was presented in topical report BAW-10173P[2], Mark-BW Reload Safety Analysis for Catawba and McGuire. BAW-10173P demonstrates that the use of Mark-BW fuel in these plants does not reduce the existing safety margin. Table 3.4-1 of BAW-10173P, entitled Input Parameters and Initial Conditions for Transients, presents a comparison of the values used in the topical report analysis and in the Catawba and McGuire FSAR analyses.

The key parameters that have the greatest effect on determining the outcome of a transient were determined in Section 6.0 of BAW-10173P. Comparisons of these key parameter values to the parameter values for Cycle 6 are shown in Table 7-1 and Figure 7-1.

The Catawba 1 Cycle 6 calculated parameters are all within the limiting values discussed in BAW-10173P. The cycle specific evaluations of several transients are presented in the following sections.

7.1.1 Steam Line Break Analysis

BAW-10173P provided a steam line break transient analysis for the offsite power available case. Revision 1[12] and Revision 2[13] of BAW-10173P provide the corresponding steam line break transient analysis for the offsite power not available case. Cycle specific statepoint analyses were performed to confirm that these analyses apply to Cycle 6. The reactivities of the statepoints were found to be less than the values used in the analyses. The minimum DNBRs of the offsite power available and offsite power not available statepoints were 1.57 and 1.88 respectively. These results verify that the existing offsite dose analysis for steam line break is applicable to Catawba 1 Cycle 6.

7.1.2 Rod Ejection

The cycle specific parameters were checked with the limiting parameters for the rod ejection transient and were acceptable. These parameters are shown in Table 7-3.

7.1.3 RCCA Misoperation

As indicated in BAW-10173P, cycle specific evaluations were performed for RCCA misoperations for dropped rods, rod misalignment, and single rod withdrawal pin census. The results of the dropped rod evaluation as described in BAW-10173P are shown in Figure 7-3. All the possible dropped rod combinations within each individual group or bank were analyzed, and the peaking for these cases was less than the DNBR peaking limit.

A statically misaligned single RCCA of control bank D could either be fully inserted or fully withdrawn. The peaking for a fully inserted RCCA is the same as the peaking for a dropped bank D RCCA and has been analyzed. A fully withdrawn single bank D RCCA with the remaining rods at the insertion limit was analyzed at BOC and EOC. The peaking increases for the fully withdrawn cases were less severe than the peaking increases for the dropped rod cases, and were therefore also acceptable.

The peaking for Catawba 1 Cycle 6 is less severe than the power distribution used to determine the number of pins failed in the single RCCA withdrawal analysis in BAW-10173P. Therefore, the number of pins failed is bounded by the value of 5 percent in BAW-10173P.

7.1.4 Locked Rotor

The locked rotor analysis in BAW-10173P was performed with a design peaking distribution that is typically conservative when compared to an actual fuel cycle design. However, the BWFC core design methodology uses maximum allowable peaking (MAP) limits to permit trade-offs between radial and axial peaks in setting AFD operating limits. These MAP limits define combinations of radial and axial peaks that maintain DNBR equivalence to the design peaking distribution. For the cycle specific evaluation of the locked rotor event, the design MAP limits (i.e. those MAP limits with DNB equivalence to the design peaking distribution) were reduced so that the allowable radial peak at the design axial (1.55 at 0.5 X/L) was equivalent to the hot pin radial peak which produced an acceptable DNBR during the locked rotor transient. This radial peaking value had been determined by evaluating the locked rotor transient at successively lower peaking values until the hot pin peak that precluded DNB was determined. To determine the percentage of pins in DNB, predicted peaking distributions for limiting operating conditions of Catawba 1, Cycle 6 were compared to the adjusted MAP limits. All pins with peaks that exceeded the adjusted MAP limits were then assumed to be in DNB.

An evaluation of the Catawba 1, Cycle 6 core design has been performed and the results indicate that the percentage of pins in DNB is less than 3.3%. This amount of pins in DNB is less than the assumption of 10% failed fuel used in the radiological consequences evaluation for the locked rotor accident. Also, since the peak clad surface temperature is less than 1800°F the core will remain in place and intact with no loss in cooling capability.

7.2 ECCS Analysis

A LOCA analysis, applicable to the Westinghouse designed nuclear plants operated by the Duke Power Company, McGuire Units 1 and 2 and Catawba Units 1 and 2, has been performed by BWFC. The analysis supports operation of the four Duke units with Mark-BW fuel, and is documented in topical report BAW-10174[14]. Methodology employed in the analysis is in accord with 10CFR50 Appendix K and is documented in topical report BAW-10168P, Revision 1[15]. The LOCA evaluation considered both large and small breaks, and transition cores containing mixed Mark-BW and OFA fuel. The evaluation concluded that the small break LOCA (SBLOCA) FSAR analyses, performed by Westinghouse, remain valid for plant licensing during the transition cycles and even after the core is loaded with BWFC-supplied fuel. It was further concluded that, under mixed core operation, the Westinghouse FSAR analysis remains valid for OFA licensing. Core LOCA limits, resulting from the evaluations presented in BAW-10174 and BAW-10174, Revision 1[16], are given in the Core Operating Limits Report for Catawba Unit 1 Cycle 6. All LOCA configurations were found to be in conformance with the five criteria of 10CFR50.46, thus demonstrating conservative results for the operation of Catawba Unit 1 Cycle 6.

7.3 Radiological Consequences

Each FSAR accident analysis has been evaluated to determine the effects of Cycle 6 operation, and to ensure that the radiological consequences of hypothetical accidents are within applicable regulatory guidelines and do not adversely affect the health and safety of the public. The design basis LOCA evaluations assessed the radiological impact of differences between the Mark-BW fuel and Westinghouse OFA fuel fission product core inventories. Also, the dose calculation effects from non-LOCA transients reanalyzed by BWFC utilizing Cycle 6 characteristics were evaluated. Differences in the current FSAR dose values that are not related to the insertion of Mark-BW fuel reflect the application of the latest revisions to Standard Review Plan dose assessment methodology. A brief discussion of each accident analyzed is provided below. A summary of the calculated radiological consequences is provided in Table 7-3. The calculated radiological consequences are all within specified regulatory guidelines and contain significant levels of margin.

7.3.1 Loss of Coolant Accidents

The offsite radiological consequences of a design basis LOCA are calculated utilizing the applicable assumptions contained within Regulatory Guide 1.4 and Standard Review Plan Sections 6.5.2, 6.5.3, 6.5.4, and 15.6.5. The control room radiological consequences are calculated utilizing the additional assumptions within Standard Review Plan Section 6.4 that are applicable to the Catawba Control Room Ventilation design. The rod ejection accident offsite dose calculation is based on assumptions provided in Regulatory Guide 1.77 and Standard Review Plan 15.4.8. Chapter 8 contains applicable FSAR Chapter 15 pages appropriately revised to reflect the assumptions used in the LOCA consequence analyses.

Using similar methodology, fission product core inventories were calculated assuming a reactor core containing Mark-BW fuel and one with Westinghouse OFA fuel. The results provided in Table 7-3 demonstrate that utilizing the Mark-BW fuel design as a replacement for the Westinghouse OFA fuel design produces differences of less than one percent in the calculated doses associated with the design basis LOCA analyses that are in the 1989 Catawba FSAR. However, there are several important assumptions which have been revised in the analyses described in this report that result in more significant differences in the current FSAR dose values.

The most important conservatism that has been added addresses the manner in which mixing and filtration are assumed to occur within the Annulus. The analyses presented in the FSAR assume mixing of Containment leakage in 50 percent of the Annulus volume prior to filtration by the Annulus Ventilation filters. The analysis presented in this report assumes that Containment leakage is processed directly by the Annulus Ventilation system filters prior to mixing in the Annulus. The net effect of this conservatism is to decrease the calculated hold-up time for radioactive decay within the Annulus and, thus, increase the calculated radioactivity releases to the environment. This assumption is consistent with Standard Review Plan Section 6.5.3.

Another important conservatism added to the offsite dose analysis affects the assumed post-accident leakage of ESF components outside Containment. The analysis contained in the FSAR assumes that the maximum operational leakage occurs throughout the accident. The analysis provided in this report also includes the leakage from a gross failure of a passive component. The leakage is conservatively assumed to be 50 gallons per minute, starting at 24 hours after the LOCA and lasting for 30 minutes. Although safety-related portions of the Auxiliary Building Ventilation System service those areas of the plant where such a gross failure is most likely to occur, no iodine removal credit is assumed for the Auxiliary Building Ventilation System filters. The net effect of this conservatism is to increase the calculated release of radioactivity to the environment. This assumption is consistent with Standard Review Plan Section 15.6.2, Appendix B.

An overly conservative assumption that has been removed deals with radioactivity release pathways. The LOCA dose analyses currently within the FSAR assume total failure of the redundant, safety-related hydrogen recombiners located in Containment. This assumption necessitates the assumed use of the Hydrogen Purge System to maintain Containment hydrogen concentrations below the theoretical flammability limit of 4 v/o. Such an assumption clearly requires two active failures, an assumption that is not consistent with other accident analyses evaluated in the FSAR. The analysis provided in this report assumes a single active failure that disables one hydrogen recombiner. The remaining operable hydrogen recombiner is able to maintain the Containment hydrogen concentration well below the 4 v/o flammability limit, thus obviating the need for the assumed use of the Hydrogen Purge System. The net effect of this assumption is to reduce the calculated release of radioactivity to the environment. This assumption is consistent with 10 CFR Part 50.44.

The total net effect of these additional assumptions when compared to the LOCA analyses provided in the 1989 FSAR is to increase the calculated whole body and skin doses, and to reduce the calculated thyroid doses. As mentioned previously, in all cases the applicable regulatory guidelines contained within 10 CFR Part 100 and General Design Criteria 19 are met with significant levels of margin.

7.3.2 Locked Rotor Accident

The radiological consequences from a reactor coolant pump rotor seizure were reanalyzed due to the results from BAW-10173P that predicted 3.3% of the pins in the core would be in DNB. Regulatory guidance given in Standard Review Plan 15.3.3 provided the basis for the offsite dose consequence assessment from a locked rotor accident. The calculated doses are presented in Table 7-3. Technical Specification limits on primary and secondary coolant activities limit the potential doses to a small fraction of the 10 CFR Part 100 exposure guidelines. The calculated doses are within 10 CFR Part 100 exposure guidelines even if the accident should occur with an iodine spike.

7.3.3 Single RCCA Withdrawal at Power

The most limiting rod cluster control assembly misoperation, accidental withdrawal of a single RCCA, is predicted to result in less than 5% fuel clad damage. The subsequent reactor and turbine trip would result in atmospheric steam dump, assuming the condenser was not available for use. The radiological consequences from this event would be less than the locked rotor event, analyzed in FSAR Section 15.3.3 and Section 7.1.4 of the Reload Report.

Table 7-1. Safety Analysis Checklist for Physics Data

Parameter	Limit Value	Cycle 6 Value
Moderator Coefficients, pcm/F		
HZP, Maximum	< +7	+4
HFP, Maximum	< .0	-2
All, Minimum	> -41	-32
Power Coefficients, pcm/%power		see figure 7-1
Beff, %		
Maximum	< .75	.63
Minimum	> .44	.52
Trip Reactivity, pcm	4000	see figure 7-2
Shutdown Margin, pcm	> 1300	2120
Maximum Differential Rod Worth, pcm/sec	< 63.75	40.3
$F_{\Delta H}$ at HZP	< 2.48	2.12

Table 7-2. Rod Ejection Parameters

Item		Limit	Cycle 6 Value
Max Ejected Rod Worth, pcm			
	BOC, HZP	≤ 750	408
	BOC, HFP	≤ 230	82
	EOC, HZP	≤ 900	544
	EOC, HFP	≤ 230	110
Max F_Q after ejection *			
	BOC, HZP	≤ 11.0	7.2
	BOC, HFP	≤ 4.5	2.8
	EOC, HZP	≤ 19.0	15.5
	EOC, HFP	≤ 5.9	3.1
Beff, %			
	BOC	> .55	.62
	EOC	> .44	.52
Pin Census, %			
	BOC, HZP	≤ 10	0
	BOC, HFP	≤ 10	<3
	EOC, HZP	≤ 10	0
	EOC, HFP	≤ 10	<3
Minimum Trip reactivity, pcm			
	BOC, HZP	≥ 2000	2400
	BOC, HFP	≥ 4000	4720
	EOC, HZP	≥ 2000	2500
	EOC, HFP	≥ 4000	4950

* F_Q prior to ejection is less than 2.32 set by the LCOs.

Table 7-3. Radiological Consequences Dose Results Summary

I. DBA Offsite Dose with ECCS Leakage Sources (Rem)

	Exclusion Area Boundary		Low Population Zone	
	Whole Body	Thyroid	Whole Body	Thyroid
A. Westinghouse OFA Core Inventory	9.08	127.0	1.14	32.1
B. Mark-BW Fuel Core Inventory	9.10	127.0	1.14	32.1

II. DBA Offsite Dose without ECCS Leakage Sources (Rem)

	Exclusion Area Boundary		Low Population Zone	
	Whole Body	Thyroid	Whole Body	Thyroid
A. Westinghouse OFA Core Inventory	9.05	118.0	1.12	13.4
B. Mark-BW Fuel Core Inventory	9.07	118.0	1.12	13.4

III. Control Room Operator Dose (Rem)

	Whole Body	Skin	Thyroid
A. Westinghouse OFA Core Inventory	1.63	32.1	14.2
B. Mark-BW Fuel Core Inventory	1.64	32.0	14.2

Table 7-3. Radiological Consequences Dose Results Summary (cont.)

IV. Rod Ejection Accident Offsite Dose (Rem)

	Exclusion Area Boundary		Low Population Zone	
	Whole Body	Thyroid	Whole Body	Thyroid
A. Westinghouse OFA Core Inventory				
Primary	2.75E-1	5.91	3.22E-2	6.72E-1
Secondary	2.20	1.77E+1	1.57E-1	5.95
B. Mark-BW Fuel Core Inventory				
Primary	2.75E-1	5.90	3.22E-2	6.71E-1
Secondary	2.20	1.76E+1	1.58E-1	5.95

V. Locked Rotor Accident Offsite Dose (Rem)

	Exclusion Area Boundary		Low Population Zone	
	Whole Body	Thyroid	Whole Body	Thyroid
1. Case 1 (No iodine spike)	4.41E-1	3.63	3.16E-2	1.20
2. Case 2 (Pre-iodine spike)	4.41E-1	3.67	3.16E-2	1.21

Figure 7-1

Doppler Power Coefficient

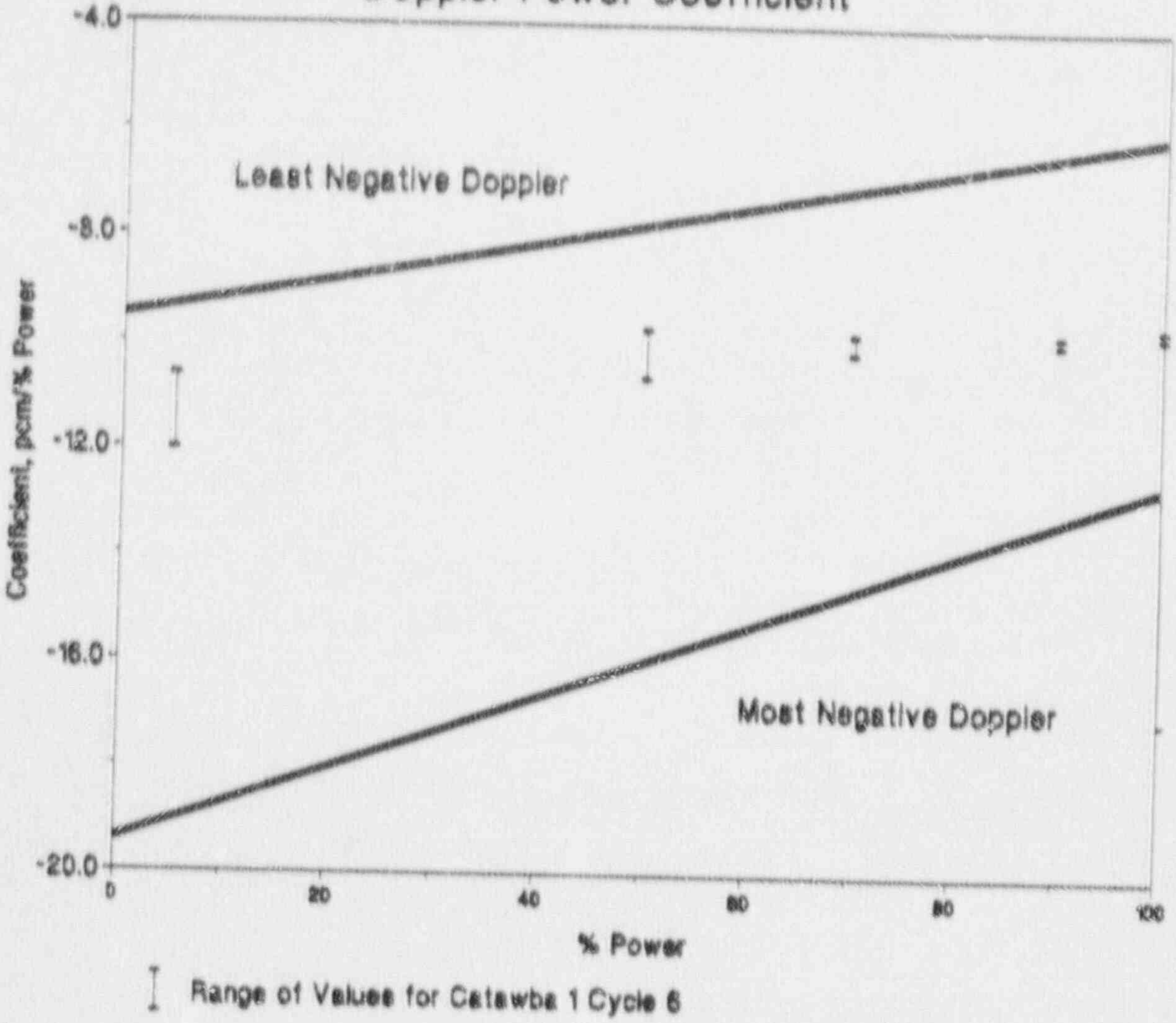


Figure 7-2

Scram Curve

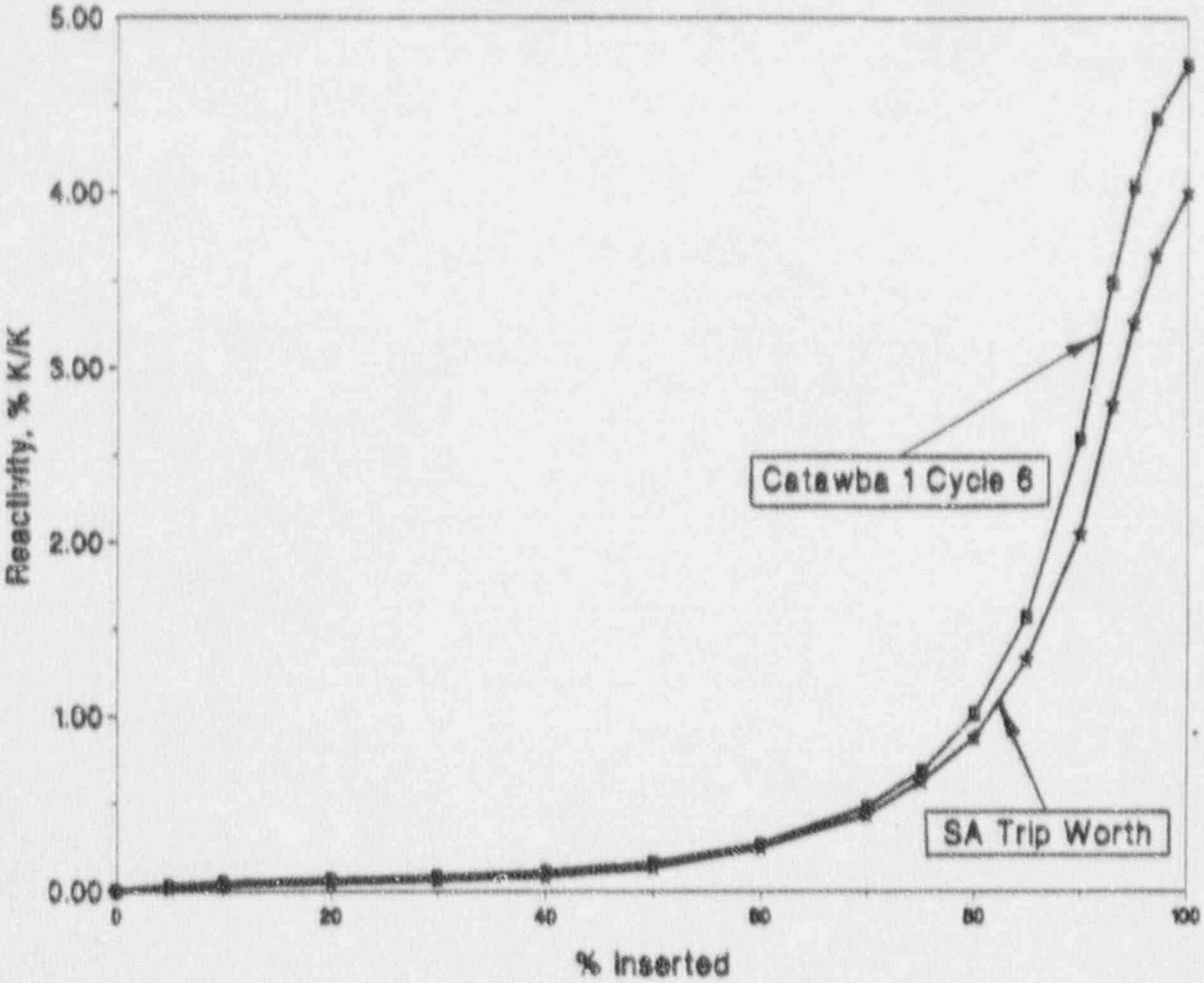
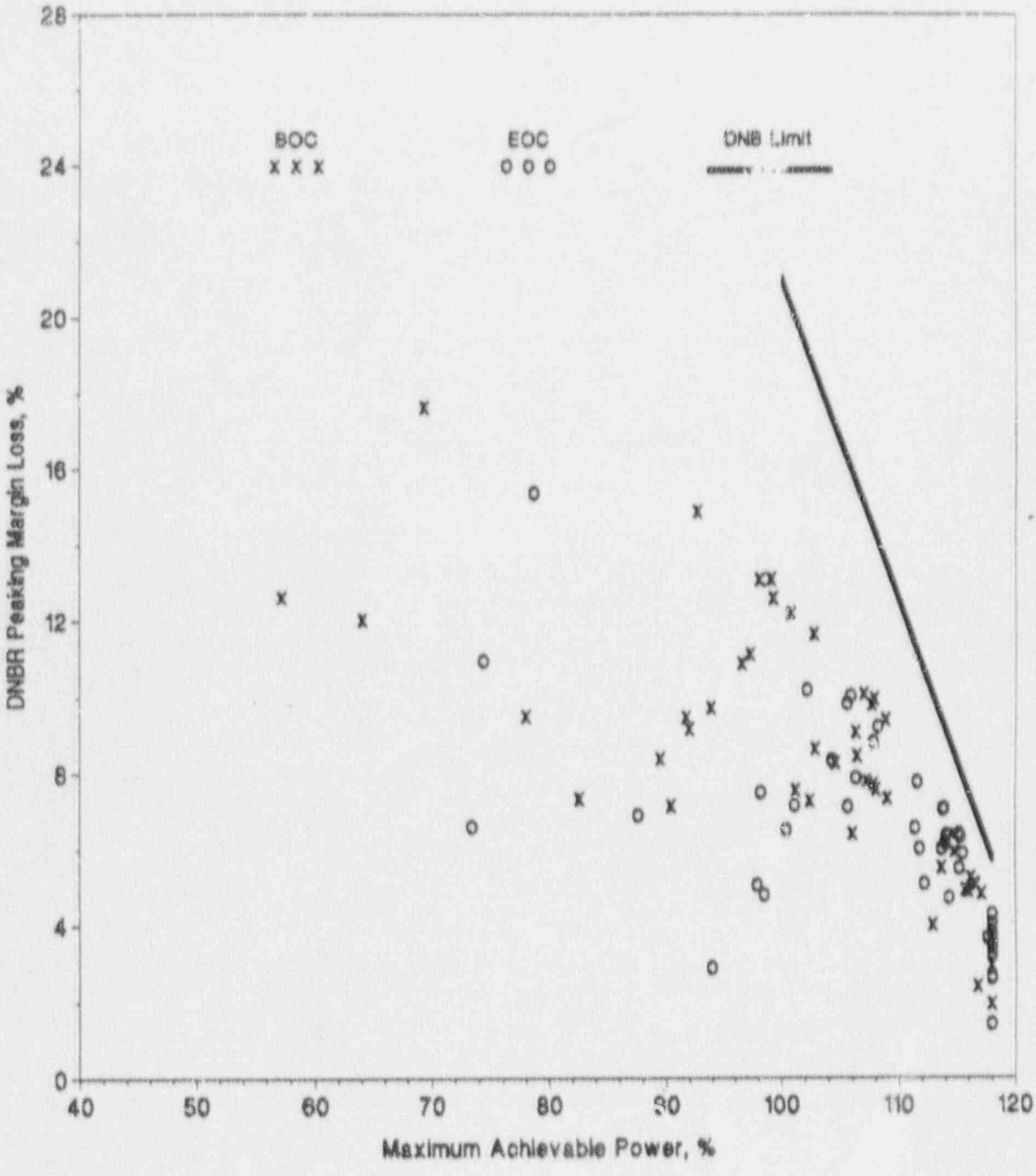


Figure 7-3
Catawba 1 Cycle 6
Dropped Rod Peaking



8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS AND COLR

The Technical Specifications and Core Operating Limits Report (COLR) have been revised for Cycle 6 operation to accommodate the influence of the Cycle 6 core design on power peaking, reactivity, and control rod worths. The Technical Specification limits and COLR limits also reflect changes in reload analysis methodology[17] beginning with this core. The Cycle 6 design analysis basis includes a low-leakage fuel cycle design and a mixed core containing both B&W Mark-BW and Westinghouse OFA fuel assemblies.

A Cycle 6-specific power distribution analysis of the final core design was conducted to generate the $f(\Delta I)$ limits for the Overpower Delta-T and Overtemperature Delta-T trip functions and the Limiting Conditions for Operation (control bank insertion and axial flux difference). The $f(\Delta I)$ limits preserve the centerline fuel melt and steady-state DNBR limits, and the Limiting Conditions for Operation preserve the maximum allowable LOCA and initial condition DNB peaking limits, ejected rod worth reactivity limits, and the shutdown margin reactivity limit. These limits were developed based on the NRC-approved methodology described in Reference 17. A peaking penalty for quadrant power tilt was taken in the analysis so that the resulting limits accommodate quadrant power tilt ratios up to a value of 1.02.

The maximum allowable LOCA peaking limits shown in Figure 4 of the COLR are based on the BWFC ECCS evaluation described in Section 7.2. A composite K(Z) limit was developed based on both large and small break analyses. Separate composite limits applicable to Mark-BW and OFA fuel were used in the power distribution analysis, and are specified in the COLR. These limits were used directly in determination of the control rod insertion and axial flux difference operating limits given in Technical Specifications 3.1.3.6 and 3.2.1. Technical Specification 3.2.2 provides the nuclear heat flux hot channel (F_Q) peaking limit.

The initial condition DNB maximum allowable peaking (MAP) limits shown in Table 3 of the COLR are based on core reference design peaking factors. The MAP limits provide allowable combinations of peaking factors that preserve DNBR performance equivalent to the design power distribution for a limiting loss of coolant flow transient. The initial condition MAPs are used as described in Reference 17 to calculate DNB peaking margins for determination of the control rod position and axial flux difference operating limits given in Technical Specifications 3.1.3.6 and 3.2.1. Technical Specification 3.2.3 provides the nuclear enthalpy rise hot channel ($F_{\Delta H}$) peaking limit.

The methodology for surveillance of the core hot channel peaking factors, $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$, is described in Reference 17. In this application of the methodology, Duke Power Company has elected to bypass the first tier surveillance (comparison of measured peaking to predicted design peaking), and to perform the peaking margin calculation directly whenever an incore flux map is taken for surveillance monitoring. This is a conservative application of the monitoring methodology and is therefore acceptable. Specifications 4.2.2 and 4.2.3 have been written in a form that provides this capability, and only the parameters required by this application of core monitoring are provided in the COLR.

The core operating limits are provided in the Core Operating Limits Report, in accordance with NRC Generic Letter 88-16 and Technical Specification 6.9.1.9. Table 8-1 lists the Technical Specification changes required for Cycle 6, while Table 8-2 lists the changes to the core operating limits contained in the COLR. These changes are being submitted to the NRC under separate covers. Parameters related to monitoring the core power distribution are defined in Reference 17, and are used by the plant computer software. These parameters will be supplied for inclusion in the COLR.

Based on the analyses and revisions to the Technical Specifications and COLR described in this report, Cycle 6 of Catawba Unit 1 will operate within the Final Acceptance Criteria ECCS limits and within the thermal design criteria. The following pages contain the required Technical Specification revisions and the revisions to the core operating limits specified in the COLR.

Table 8-1. Technical Specifications Changes

Applicable Tech. Spec. No.	Reason for Change
2.1.1	changed CHF correlation increased $F_{\Delta H}$ for Mark-BW fuel reduced minimum measured RCS flow
2.2.1	completed RTD bypass removal reduced minimum measured RCS flow increased error allowances on certain reactor trip instrumentation
3/4.2.1	deleted baseload operation
3/4.2.2	changed F_Q methodology
3/4.2.3	changed $F_{\Delta H}$ methodology separated RCS flow and $F_{\Delta H}$ relationship
3/4.2.4	increased the tilt ratio at which a power reduction is required rewrote LCO to be consistent with Westinghouse STS
3/4.2.5	incorporated RCS flow as a DNB parameter removed power/flow tradeoff dependence on R reduced minimum measured RCS flow
4.5.2	changed flow and developed pressure requirements to be consistent with revised accident analysis flow assumptions
6.9.1.9	reflected the change to BWFC operating limit methodology

Table 8-2. Core Operating Limits Report Changes

Applicable Tech. Spec. No.	Reason for Change
3/4.1.3.5	revised safety bank insertion limits to reflect a rod withdrawal limit of 222 steps
3/4.1.3.6	revised control bank insertion limits to reflect a rod withdrawal limit of 222 steps
3/4.2.1	revised AFD limits for Cycle 6 operation
3/4.2.2	revised for Cycle 6 operation to reflect a change in the Heat Flux Hot Channel factor $F_Q(X,Y,Z)$ methodology
3/4.2.3	revised for Cycle 6 operation to reflect a change in the Nuclear Enthalpy Rise Hot Channel factor $F_{\Delta H}(X,Y)$ methodology

Changes to Technical Specifications

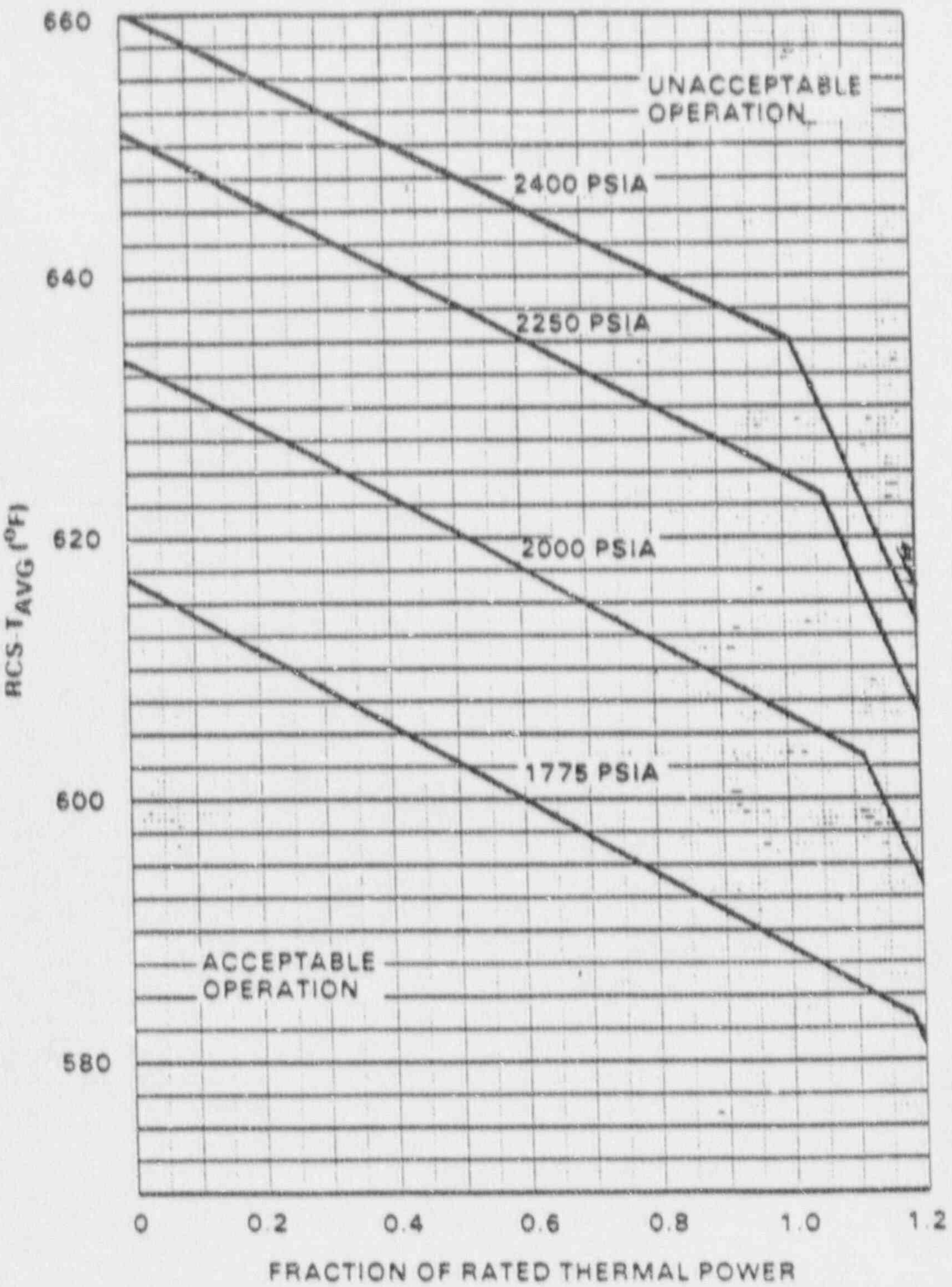


FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

Figure 2.1-1 Reactor Core Safety Limits - Four Loops in Operation

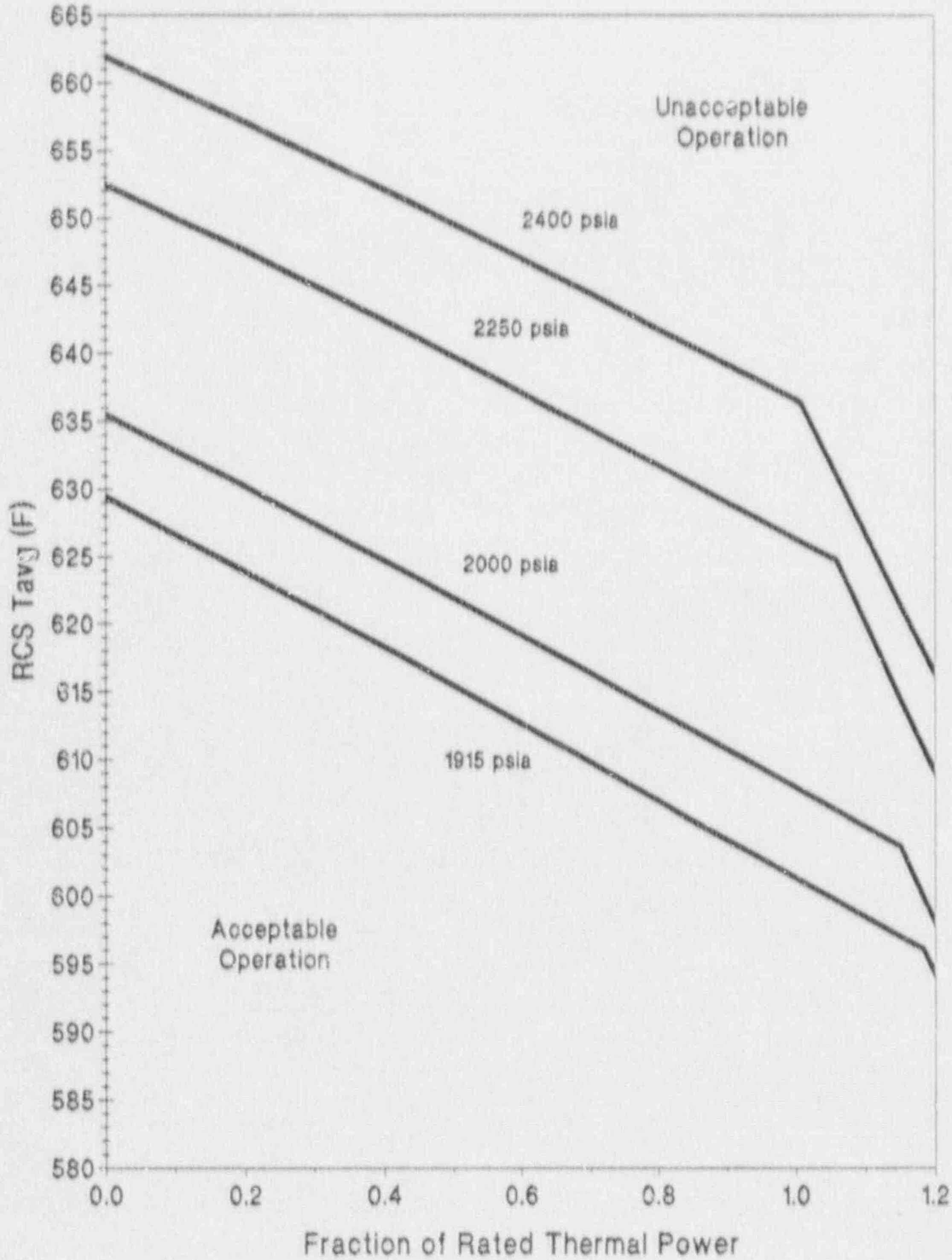


TABLE 2.2.-1

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~~ 5.92	0	≤109% of RTP*	≤~~111.1%~~ 110.9 of RTP*
b. Low Setpoint	8.3	~~4.56~~ 5.92	0	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥ 2 seconds	≤6.3% of RTP* with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP*	≤31% of RTP*
6. Source Range, Neutron Flux	17.0	10	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT	~~7.2{8.9#}~~ 6.98	~~4.47{5.41#}~~ 3.0#	~~2.03{2.65#}~~ 2.12#	See Note 1	See Note 2
8. Overpower ΔT	~~4.3{4.9#}~~	~~1.3{1.24#}~~	~~1.2{1.7#}~~	See Note 3	See Note 4
9. Pressurizer Pressure-Low	4.0	2.21	1.5	≥1945 psig	≥1938 psig***
10. Pressurizer Pressure-High	7.5	~~4.96~~ 0.71	~~0.5~~ 1.5	≤2385 psig	≤2399 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span
12. Reactor Coolant Flow-Low	~~2.5~~ 2.92	~~1.77(1.41#)~~ 1.48	0.6	≥90% of loop minimum measured flow**	≥~~89.2% (88.8%#)~~ 88.9 of loop minimum measured flow**

*RTP = RATED THERMAL POWER

**Loop minimum measured flow = ~~96,900~~ 96,250 gpm

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

\# [illegible]licable upon deletion of RTD Bypass System

~~Amendment No. 42 (Unit 1)~~
~~Amendment No. 35 (Unit 2)~~

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT = Measured ΔT by RTD Manifold Instrumentation;

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

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1 = $\cancel{8}$ 12* s, τ_2 = 3 s;

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

τ_3 = Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

K_1 = $\cancel{1.4113}$ 1.38*;

K_2 = 0.02401/°F;

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

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = $\cancel{28}$ 22* s, τ_5 = 4 s;

T = Average temperature, °F;

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

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$;

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

NOTE 1: (Continued)

T'	$\leq$	590.8°F (Nominal T_{avg} allowed by Safety Analysis);
K_3	=	0.001189;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

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

(i) For $q_t - q_b$ between -22.5% and -6.5%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent that the magnitude of $q_t - q_b$ is more negative than -22.5%, the ΔI Trip Setpoint shall be automatically reduced by 3.151% of its value at RATED THERMAL POWER; and

(iii) For each percent that the magnitude of $q_t - q_b$ is more positive than -6.5%, the ΔI Trip Setpoint shall be automatically reduced by 1.641% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~2.4%~~ 3.0%.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

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_0 = As defined in Note 1,

K_4 = 1.0704,

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 controller for T_{avg} dynamic compensation,

τ_7 = Time constant utilized in the rate-lag controller 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 (Continued)

NOTE 3: (Continued)

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

T = As defined in Note 1,

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

S = As defined in Note 1, and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~2.6%#~~ 2.8%.

~~#Applicable upon deletion of RTD Bypass System~~

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 through the ~~WRB-1~~ correlation. The ~~WRB-1~~ DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio, (DNBR), is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

BWCMV

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

, and in the BWCMV DNB 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% confidence that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above ~~plant~~ parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty~~, combined with the correlation DNBR limit,~~ is used to establish~~es~~ a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

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

2.1 SAFETY LIMITS

BASES

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

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

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

Where P is the fraction of RATED THERMAL POWER.

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

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

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

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

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

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

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

Delete

a. the allowed operational space as specified in the CORE OPERATING LIMITS REPORT (COLR) for RAOC operation, or

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

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

ACTION:

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

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

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

Delete

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

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

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

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

Delete

*See Special Test Exceptions Specification 3.10.2.

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

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

c. The provisions of Specification 4.0.4 are not applicable.

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

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

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

Delete

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

LIMITING CONDITION FOR OPERATION

$F_Q(X,Y,Z)$

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

$F_Q^{MA}(X,Y,Z)$

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

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

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

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

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

APPLICABILITY: MODE 1.

ACTION:

Replace with Attachment 1

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

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit, and

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

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

Attachment 1:

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

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

c. POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit, and

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 ~~For RAOC operation,~~ $F_Q(z)$ $F_Q^M(x,y,z)^{(1)}$ shall be evaluated to determine whether ~~if~~ $F_Q(z)$ $F_Q(x,y,z)$ is within its limit by:

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

Delete

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

c. Satisfying the following relationship:

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

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

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(z), and W(z) are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

Replace with Attachment 2

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.

Attachment 2:

b. Measuring $F_Q^M(X,Y,Z)$ at the earliest of:

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

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

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

---

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

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

e. With measurements indicating

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

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

Replace with Attachment 4

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

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

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

f. With the relationships specified in Specification 4.2.2.2c. above not being satisfied:

Replace with Attachment 3

1) Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

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

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

2) One of the following actions shall be taken:

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

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

Delete

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

Attachment 3:

c. Performing the following calculations:

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

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

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

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

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

(a) Within 15 minutes:

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

$$(\text{AFD Limit})^{\text{reduced}}_{\text{negative}} = (\text{AFD Limit})^{\text{COLR}^{(3)}}_{\text{negative}} + [\text{NSLOPE}_1{}^{(3)} \times \text{Margin}^{\text{min}}_{\text{OP}}]\ \text{absolute value}$$

$$(\text{AFD Limit})^{\text{reduced}}_{\text{positive}} = (\text{AFD Limit})^{\text{COLR}^{(3)}}_{\text{positive}} - [\text{PSLOPE}_1{}^{(3)} \times \text{Margin}^{\text{min}}_{\text{OP}}]\ \text{absolute value}$$

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

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

(b) Comply with the ACTION requirements of Specification 3.2.2, treating the margin violation in 4.2.2.2.c.1 above as the amount by which F_Q^{MA} is exceeding its limit.

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

Attachment 3 (con't):

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

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

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

where $MARGIN_{RPS}^{min}$ is the minimum margin from 4.2.2.2.c.1.

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

(4) K_1 value from Table 2.2-1.

Attachment 4:

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

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

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

either of the following actions shall be taken:

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

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

SURVEILLANCE REQUIREMENTS (Continued)

Replace with Attachment 5

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.

Delete

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

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within the target band about the target flux difference of Specification 3.2.1) during this time period. Base Load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and FQ surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

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

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(Z)$ as a function of core height. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during Base Load operation. F_Q^{RTP}, K(z), and $W(Z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

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

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

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

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

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

Attachment 5:

e. The limits in Specifications 4.2.2.2.c and 4.2.2.2.d are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

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

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

SURVEILLANCE REQUIREMENTS (Continued)

Delete

c. Satisfying the following relationship:

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

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. F_Q^{RTP} is the F_Q limit. $K(Z)$ is the normalized $F_Q(Z)$ as a function of core height. P is the relative THERMAL POWER. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during Base Load operation. F_Q^{RTP}, $K(Z)$, and $W(Z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

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

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

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

e. With measurements indicating

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

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

Delete

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

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

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

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

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

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

Replace with Attachment 6

Attachment 6:

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

SURVEILLANCE REQUIREMENTS

4.2.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

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

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties,

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

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

2) The relationship:

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

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

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

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

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

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

Delete

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

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

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

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

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

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

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

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

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

Delete

3/4.2.3 ~~REACTOR COOLANT SYSTEM FLOW RATE AND~~ NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

Replace with Attachment 1

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

Where:

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

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

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

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

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

APPLICABILITY: MODE 1.

ACTION:

Replace with Attachment 2

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

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

1. Within 2 hours either:

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

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

Attachment 1:

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

$F\Delta HR^{M}(X,Y) \leq F\Delta HR^{L}(X,Y)$

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

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

Attachment 2:

ACTION:

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

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

b. Within 6 hours either:

   1. Restore $F\Delta HR^M(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

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

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

   1. Restore $F\Delta HR^M(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

   2. Perform the following actions:

      (a) Reduce the $OT\Delta T$ K_1 term in Table 2.2-1 by at least TRH[2] for each 1% that $F\Delta HR^M(X,Y)$ exceeds the limit, and

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

---

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

[2] TRH is the amount of $OT\Delta T$ K_1 setpoint reduction required to compensate for each 1% that $F\Delta HR^M(X,Y)$ exceeds the limit of Specification 3.2.3, provided in the COLR per Specification 6.9.1.9.

3/4.2.3 ~~REACTOR COOLANT SYSTEM FLOW RATE AND~~ NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

Incorporated in Specification 3.2.5

ACTION (Continued)

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

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

Delete

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

a. and/or c.2.

d~~f~~. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION b.1.c) and/or b.2., above; subsequent POWER OPERATION may proceed provided that $F\Delta HR^{M}(X,Y)$ is ~~the combination of R and indicated Reactor Coolant System total flow rate are~~ demonstrated, through incore flux mapping ~~and Reactor Coolant System total flow rate comparison~~, to be within the limit ~~regions of restricted or permissible operation~~ specified in the COLR prior to exceeding the following THERMAL POWER levels:

1 ~~a~~) ~~A nominal~~ 50% of RATED THERMAL POWER,

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

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

Insert Attachment 4

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

Replace with Attachment 3

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

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

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

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

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

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

Incorporated in Specification 4.2.5

Insert Attachment 5

Attachment 3:

b. Measuring $F\Delta HR^{M}(X,Y)$ according to the following schedule:

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

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

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

Attachment 4:

4.2.3.2 $F\Delta HR^{M}(X,Y)$ shall be evaluated to determine whether $F_{\Delta H}(X,Y)$ is within its limit by:

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

Attachment 5:

c. Performing the following calculations:

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

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

No additional uncertainties are required for $F\Delta HR^{M}(X,Y)$, because $F\Delta HR^{L}(X,Y)$ includes uncertainties.

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

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

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

either of the following actions shall be taken:

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

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

POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02. ~~above 50% of RATED THERMAL POWER.~~

APPLICABILITY: MODE 1*,**

ACTION:

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

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

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

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

2. Within 2 hours either:

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

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

1.02

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

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

---

*See Special Test Exceptions Specification 3.10.2.

Insert Attachment 1

Attachment 1:

**Not applicable until calibration of the excore detectors is completed subsequent to refueling.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

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

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

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

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

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

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

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

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

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

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

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

c. The provisions of Specification 4.0.4 are not applicable.

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

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg} ~~and~~

b. Pressurizer Pressure,

c. Reactor Coolant System Total Flow Rate.

APPLICABILITY: MODE 1.

ACTION:

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

Add Insert ①, attached

SURVEILLANCE REQUIREMENTS

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

Add Insert ②, attached

Insert ①

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

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

1. Within 2 hours either:

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

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

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

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

Insert ②

4.2.~~3.4~~ 5.2 The Reactor Coolant System total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

4.2.~~3.5~~ 5.3 The Reactor Coolant System total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

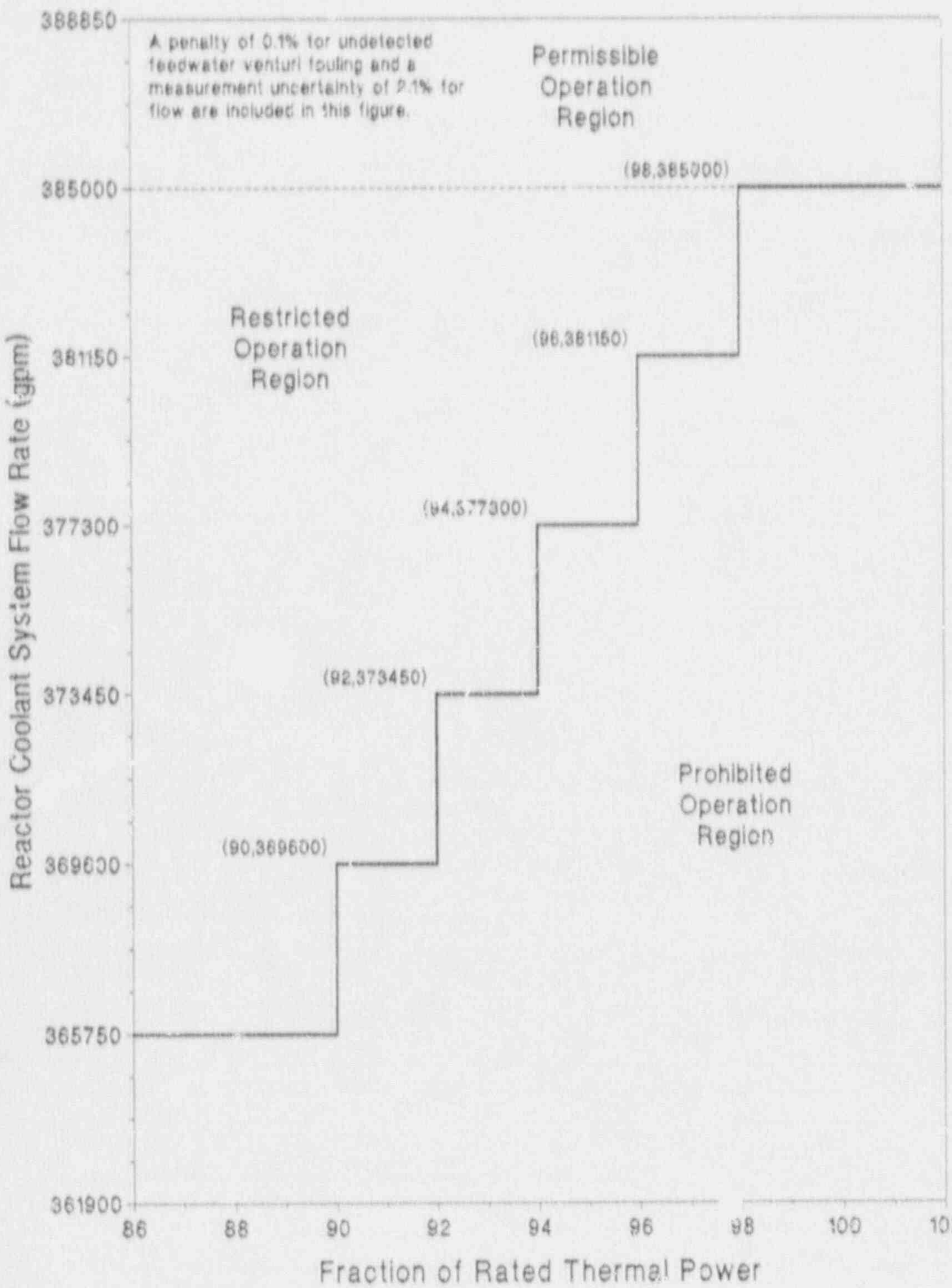
TABLE 3.2-1

DNB PARAMETERS

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

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

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



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

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

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to ~~320~~ 345 gpm, and

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

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

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to ~~462~~ 450 gpm, and

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

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

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 calculated DNBR in the core greater than or equal to design limit DNBR during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria are ~~limit of 2200°F is~~ not exceeded.

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

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

$F^N_{\Delta H}$ $F_{\Delta H}(X,Y)$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along ~~the~~ a rod ~~with the highest integrated power~~ at core location X, Y to the average rod power.

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

3/4.2.1 AXIAL FLUX DIFFERENCE

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

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

Delete

The AFD envelope specified in the COLR has been adjusted for measurement uncertainty.

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

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

Delete

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ~~(Continued)~~

Delete

the indicated AFD to relatively small target band and power swings (AFD target band as specified in the COLR, $APL^{ND} \leq$ power $\leq APL^{BL}$ or 100% Rated Thermal Power, whichever is lower). For Base Load operation, it is expected that the Units will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24 hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the Base Load procedure. After the waiting period extended Base Load operation is permissible.

The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are: 1) outside the allowed ΔI power operating space (for RAOC operation), or 2) outside the allowed ΔI target band (for Base Load operation). These alarms are active when power is greater than: 1) 50% of RATED THERMAL POWER (for RAOC operation), or 2) APL^{ND} (for Base Load operation). Penalty deviation minutes for Base Load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed.

The limits on heat flux hot channel factor, ~~coolant flow rate,~~ and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the ~~peak fuel clad temperature will not exceed the 2200°F~~ ECCS acceptance criteria are not exceeded. ~~limit.~~ ~~These~~ The peaking limits are specified in the CORE OPERATING LIMITS REPORT (COLR) per Specification 6.9.1.9.

The heat flux hot channel factor and nuclear enthalpy rise hot channel factor are each ~~Each of these is~~ measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

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

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

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

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

Incorporated in Bases for Specification 3.2.5

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

$F_{\Delta H}(X,Y)$

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on the figure specified in the CORE OPERATING LIMITS REPORT (COLR), Reactor Coolant System flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i. , a low measured Reactor Coolant System flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

Replace ... ment 1

R as calculated in Specification 3.2.3 and used in the figure specified in the COLR, accounts for $F^N_{\Delta H}$ less than or equal to the $F^{RTP}_{\Delta H}$ limit specified in the COLR. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed. The rod bow penalty as a function of burnup applied for $F^N_{\Delta H}$ is alculated with the methods described in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," July 1979, and the maximum rod bow penalty is 2.7% DNBR. Since the safety analysis is performed with plant-specific safety DNBR limits compared to the design DNBR limits. there is sufficient thermal margin available to offset the rod bow penalty of 2.7% DNBR.

The hot channel factor $F^M_Q(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either RAOC or Base Load operation, W(z) or $W(z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{BL}$ accounts for the more restrictive operating limits allowed by Base Load operation which result in less severe transient values. The W(z) function for normal operation and the $W(Z)_{BL}$ function for Base Load Operation are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

Replace with Attachment 2

Attachment 1:

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

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

where k = power factor multiplier applied to the MAP limits

P = THERMAL POWER / RATED THERMAL POWER

RRH is given in the COLR

Attachment 2:

$F_Q^M(X,Y,Z)$ and $F\Delta HR^M(X,Y)$ are measured periodically, and comparisons to the allowable limit are made to provide reasonable assurance that the limiting criteria will not be exceeded for operation within the Technical Specification limits of Sections 2.2 (Limiting Safety Systems Settings), 3.1.3 (Movable Control Assemblies), 3.2.1 (Axial Flux Difference), and 3.2.4 (Quadrant Power Tilt Ratio). A peaking margin calculation is performed to provide a basis for decreasing the width of the AFD limits, for reducing the K_1 value from OTΔT, and for reducing THERMAL POWER.

When an $F_Q^M(X,Y,Z)$ measurement is obtained in accordance with the surveillance requirements of Specification 4.2.2, no uncertainties are applied to the measured peak; the required uncertainties are included in the peaking limit. When $F_Q^M(X,Y,Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2, the measured peak is increased by the radial-local peaking factor to convert it to a local peak. Allowances of 5% for measurement uncertainty and 3% for manufacturing tolerances are then applied to the measured peak.

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

Incorporated in Bases for Specification 3.2.5

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

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

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

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

3/4.2.4 QUADRANT POWER TILT RATIO

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.

Replace with Attachment 3

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

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

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

Attachment 3:

A peaking increase that reflects a QUADRANT POWER TILT RATIO of 1.02 is included in the generation of the AFD limits.

QUADRANT POWER TILT RATIO (Continued)

Delete

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

3/4.2.5 DNB PARAMETERS [Revise Text as Indicated]

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a design limit DNBR throughout each analyzed transient. Add Insert 1 ① The indicated T_{avg} value and the indicated pressurizer pressure value correspond to analytical limits of 594.8°F and 2205.3 psig respectively, with allowance for measurement uncertainty. ② Add Insert 2, Attached

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

① Insert 1

~~$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained.~~ As noted on ~~the figure specified in the CORE OPERATING LIMITS REPORT (COLR),~~ Figure 3.2-1 Reactor Coolant System flow rate and ~~$F^N_{\Delta H}$~~ THERMAL POWER may be "traded off" against one another (i.e., a low measured Reactor Coolant System flow rate is acceptable if the ~~measured $F^N_{\Delta H}$~~ THERMAL POWER is also low) to ensure that the calculated DNBR will not be below the design DNBR value. ~~The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.~~ The relationship defined on Figure 3.2-1 remains valid as long as the limits placed on the nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, in Specification 3.2.3 are maintained.

② Insert text as indicated on next page

Insert 2
Insert at ② on preceding page.

When Reactor Coolant System flow rate ~~and $F^N_{\Delta H}$~~ is ~~are~~ measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-1 since a ~~the figure specified in the COLR.~~ Measurement error ~~s~~ of 2.1% for Reactor Coolant System total flow rate ~~and 4% for $F^N_{\Delta H}$~~ has ~~have~~ been allowed for in determination of the design DNBR value.

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

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555, with a copy to the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

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

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

3. Control Bank Insertion Limits for Specification 3/4.1.3.6,

4. Axial Flux Difference Limits, target band,* and APL^{ND}* for Specification 3/4.2.1,

5. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), W(Z),** APL^{ND}** and $W(Z)_{BL}$ for Specification 3/4.2.2, and

6. Nuclear Enthalpy Rise Hot Channel Factor, FΔHR^{L}*** or $F_{\Delta H}^{RTP}$****, and Power Factor Multiplier, $MF_{\Delta H}$,**** limits for Specification 3/4.2.3.

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

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

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

Insert Attachment 1

Attachment 1:

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

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

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

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

CORE OPERATING LIMITS REPORT (Continued)

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

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

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

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

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

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

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

Insert Attachment 2

Attachment 2:

4. BAW-10152-A, "NOODLE - A Multi-Dimensional Two-Group Reactor Simulator," June 1985.
(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

5. BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse-Designed PWR's," June 1989.
(Methodology for Specifications 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

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

PRELIMINARY

Changes to Core Operating Limits Report

2.2 Shutdown Rod Insertion Limit (Specification 3/4.1.3.5)

2.2.1 The shutdown rods shall be withdrawn to at least 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.

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

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

PRELIMINARY

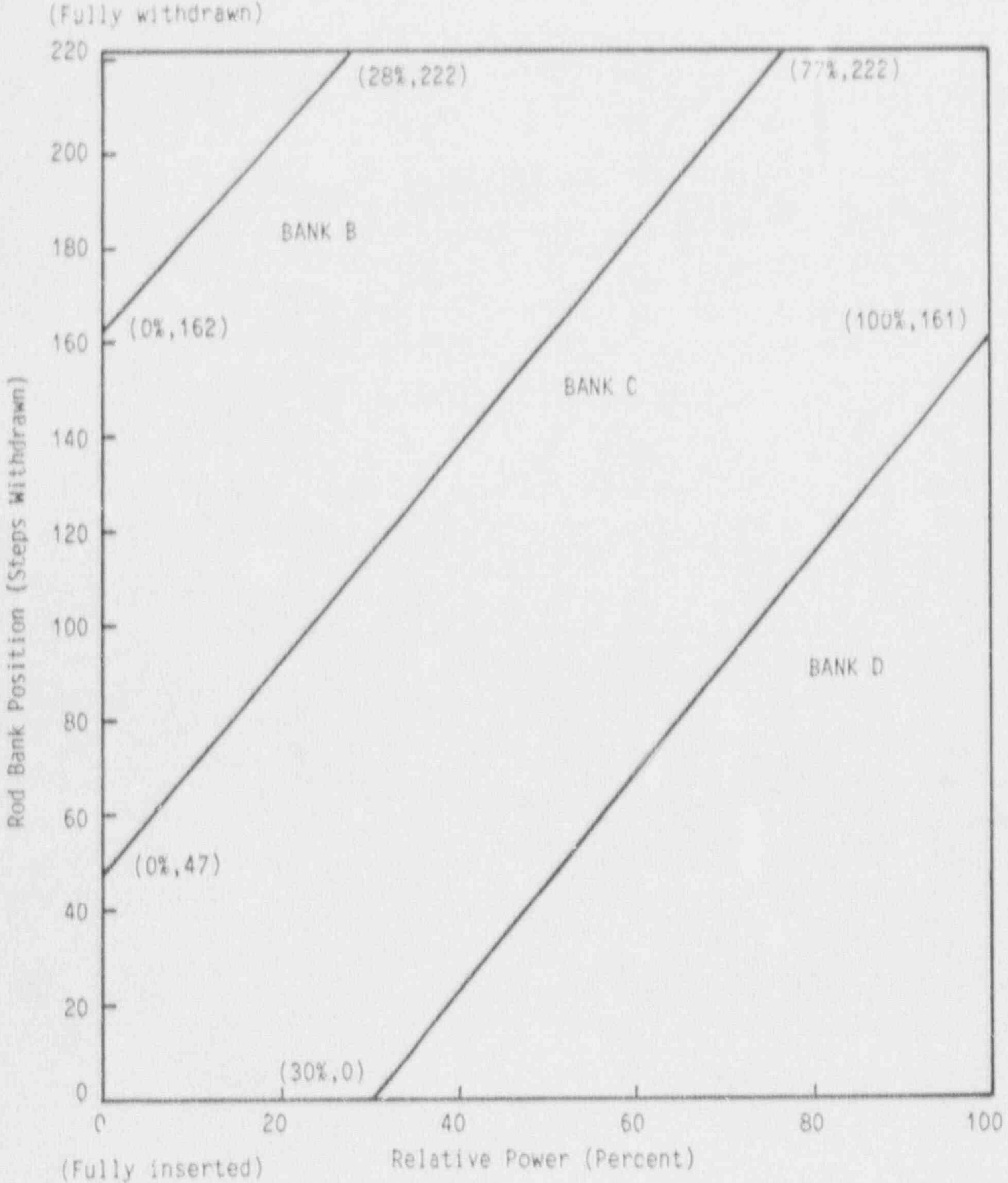

Figure 2

Control Rod Bank Insertion Limits
vs. percent RATED THERMAL POWER

PRELIMINARY

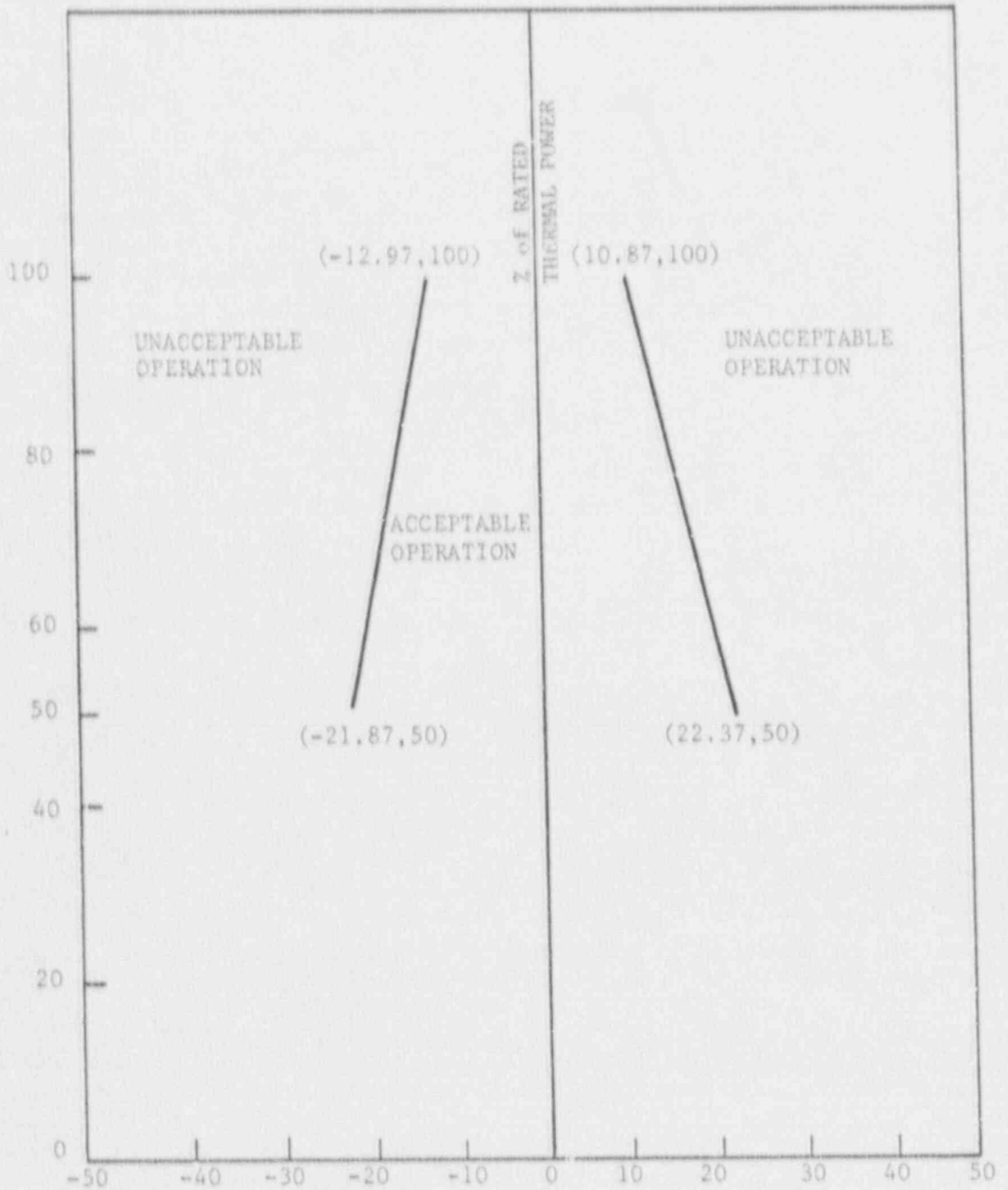

FIGURE 3

Axial Flux Difference Setpoints As A Function of RATED THERMAL POWER

PRELIMINARY

2.5 Heat Flux Hot Channel Factor (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 $[F_Q^L(X,Y,Z)]^{OP}$ is provided in Table 1.

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

Note: $[F_Q^L(X,Y,Z)]^{OP}$ is the parameter identified as B.DES in BAW-10163P-A.

2.5.5 $[F_Q^L(X,Y,Z)]^{RPS}$ is provided in Table 2.

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

Note: $[F_Q^L(X,Y,Z)]^{RPS}$ is the parameter identified as BCDES in BAW-10163P-A.

PRELIMINARY

2.5.6 $NSLOPE_i = 1.0^*$

where $NSLOPE_i$ = Negative AFD limit adjustment required to compensate for each 1% that $F_Q(X,Y,Z)$ exceeds its limit.

2.5.7 $PSLOPE_i = 1.0^*$

where $PSLOPE_i$ = Positive AFD limit adjustment required to compensate for each 1% that $F_Q(X,Y,Z)$ exceeds its limit.

2.5.8 $KSLOPE = .045^*$

where KSLOPE = Adjustment to the K_1 value from OTΔT required to compensate for each 1% that $F_Q(X,Y,Z)$ exceeds its limit.

*typical value; actual values will be supplied when monitoring inputs are computed.

PRELIMINARY

Catawba 1 Cycle 6 Core Operating Limits Report

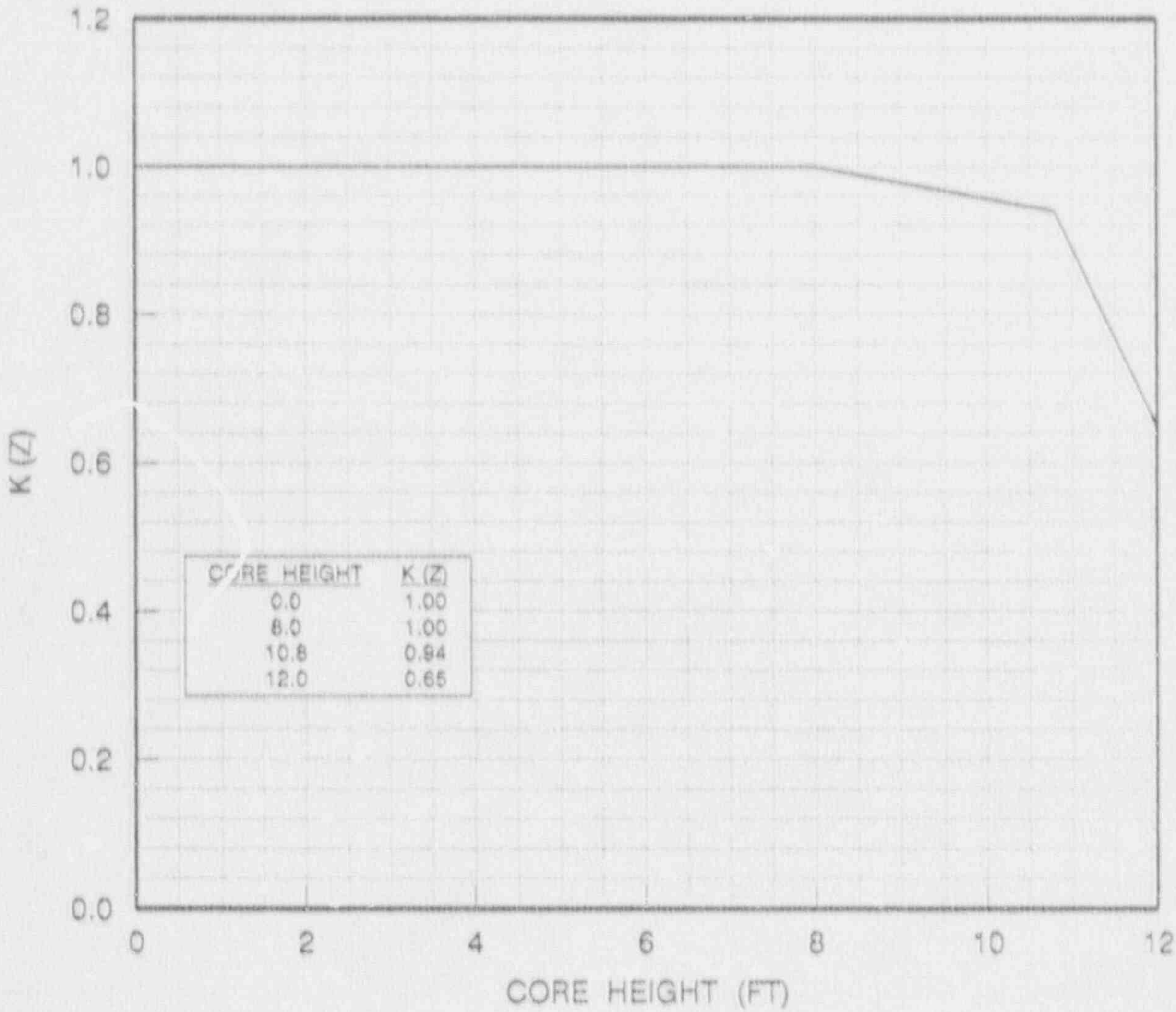

Figure 4

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

PRELIMINARY

Catawba 1 Cycle 6 Core Operating Limits Report

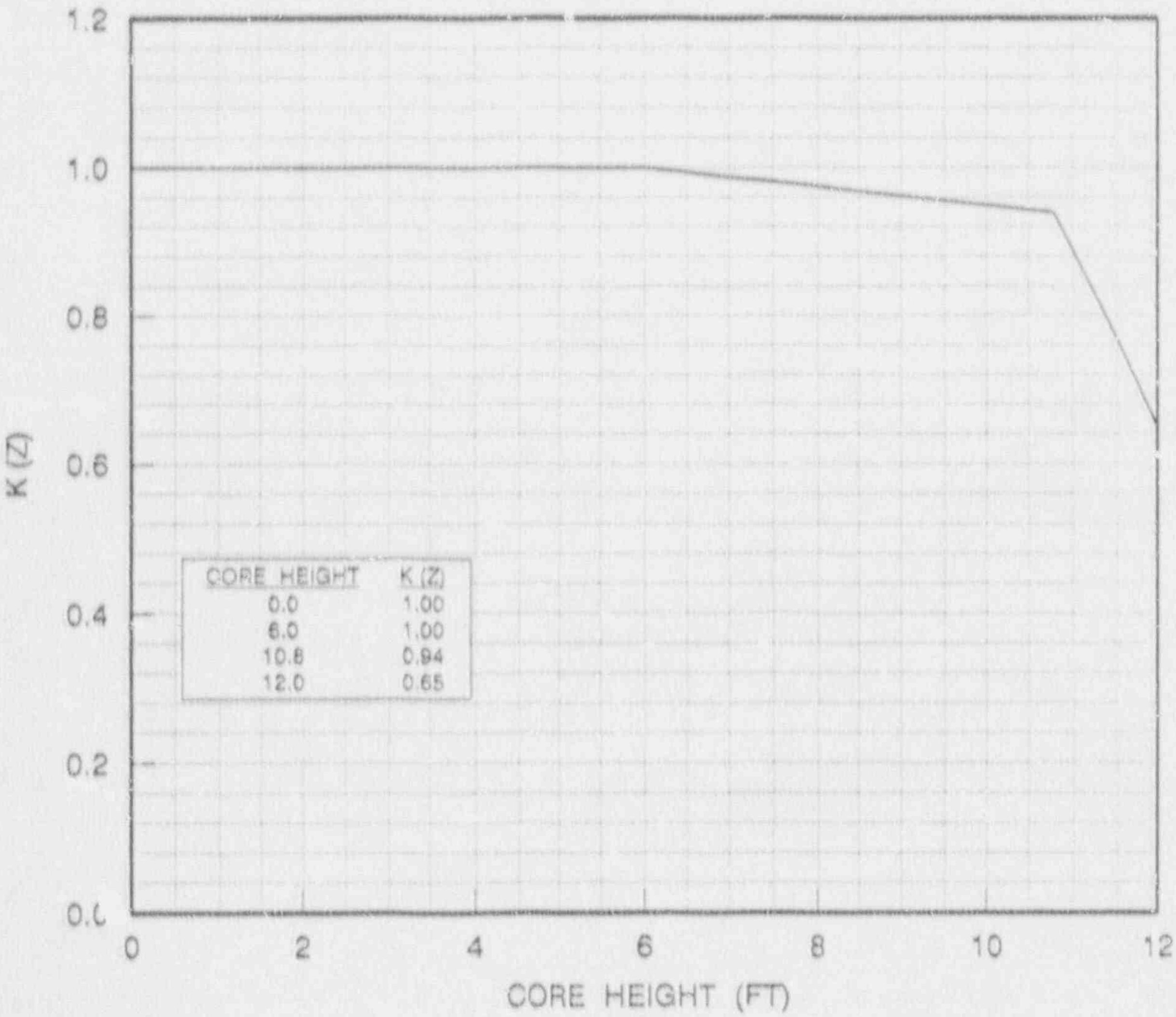

Figure 5

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

PRELIMINARY

Catawba Unit 1 Core Operating Limits Report (Continued)

2.6 <u>Nuclear Enthalpy Rise Hot Channel Factor</u> (Specification 3/4.2.3)

$F_{\Delta H}^{L}$ = MAP(X,Y,Z)/AXIAL(X,Y)

where AXIAL(X,Y) is the axial peak from the normalized axial power shape

2.6.1 MAP(X,Y,Z) is provided in Table 3.

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

2.6.2 $F\Delta HR^{L}$(X,Y) is provided in Table 4.

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

Note: $F\Delta HR^{L}$(X,Y) is the parameter identified as BHDES in BAW-10163P-A.

2.6.3 $F\Delta HR^{M}$(X,Y) = $F_{\Delta H}^{M}$(X,Y)/MAP^{M}/AXIAL(X,Y)

where $F_{\Delta H}^{M}$(X,Y) is the measured radial peak at location X,Y

MAP^{M} is the value of MAP(X,Y,Z) obtained from Table 3 for the measured peak.

2.6.4 RRH = 3.34* when $0.8 < P \leq 1.0$
RRH = 1.67* when $P \leq 0.8$

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

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

2.6.5 TRH = 0.01*

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

*typical value; actual values will be supplied when monitoring inputs are computed.

NOTE: Tables 1, 2, and 4 will be supplied when monitoring inputs are computed.

PRELIMINARY

Table 1. $[F_Q^L(X,Y,Z)]^{OP}$

(Later)

Table 2. $[F_Q^L(X,Y,Z)]^{RPS}$

(Later)

PRELIMINARY

Table 3. Catawba 1 Cycle 6 Operating Limit Maximum Allowable Total Peaks

Elevation ft.	Axial	Peak MAP(X,Y,Z) (OFA)	Peak MAP(X,Y,Z) (Mark-BW)
2	1.1	1.747	1.818
4		1.743	1.814
6		1.737	1.808
8		1.725	1.796
10		1.701	1.771
2	1.2	1.948	2.028
4		1.939	2.018
6		1.924	2.003
8		1.902	1.980
10		1.848	1.923
2	1.3	2.158	2.246
4		2.141	2.228
6		2.115	2.201
8		2.072	2.157
10		1.962	2.042
2	1.4	2.333	2.428
4		2.327	2.422
6		2.294	2.388
8		2.185	2.274
10		2.058	2.142
2	1.5	2.498	2.600
4		2.496	2.598
6		2.399	2.497
8		2.278	2.371
10		2.149	2.237
2	1.7	2.824	2.939
4		2.710	2.820
6		2.574	2.679
8		2.443	2.543
10		2.313	2.407
2	1.9	2.964	3.085
4		2.854	2.970
6		2.723	2.834
8		2.591	2.697
10		2.462	2.562

PRELIMINARY

Table 4. $F\Delta HR^{L}(X,Y)$

(Later)

Changes to Final Safety Analysis Report

15.1.2 Feedwater System malfunction causing an increase in feedwater flow

15.1.3 Excessive increase in secondary steam flow

15.1.5 Steam system piping failure

15.4.2 Uncontrolled rod cluster control assembly bank withdrawal at power

15.4.4 Startup of an inactive loop

15.5.1 Inadvertent operation of Emergency Core Cooling System during power operation

15.6.1 Inadvertent opening of a pressurizer safety or relief valve.

Loss of main feedwater flow is a Condition II occurrence by itself and is analyzed in Section 15.2.7. There is no credible reason for any of the Condition II events listed above to cause a loss of feedwater flow. Therefore, a loss of feedwater is not considered coincidently with those occurances listed above which are Condition II.

For the steamline break transient it is conservative to assume main feedwater is available. This maximizes the amount of steam generator inventory available to be blowdown and prolongs the transient.

The response times and discharge rates for some important plant valves and pumps are listed in Table 15.0.8-3.

15.0.9 FISSION PRODUCT INVENTORIES

15.0.9.1 Inventory in the Core

The fission product radiation sources considered to be released from the fuel to the containment following a maximum credible accident are based on the assumptions stated in TID-14844 (Reference 1), namely 100 percent of the noble gases, 50 percent of the halogens and a core power level of 3565 MWt.

The time-dependent fission product inventories in the reactor core are calculated ~~by the ORIGEN Code~~ (Reference 2) using a data library based on ENDF/B-IV (Reference 3). The core inventories are shown in Table 15.0.9-1.
from information given by the PDQ-XO Code

The Equilibrium Appearance Rate of Iodines in the RCS due to conservative and realistic fuel defects are shown in Table 15.0.9-2.

15.0.9.2 Inventory in the Fuel Pellet Clad Gap

The radiation sources associated with accidents which may cause more than 1 percent failed fuel (loss of coolant accident, rod cluster control assembly ejection, and fuel handing accidents) are based on the assumption that the fission products in the gap between the fuel pellets and the cladding of the damaged fuel rods are released as a result of cladding failure.

The gap activities were determined using the model suggested in Regulatory Guide 1.25. Specifically, 10 percent of the iodine and noble gas activity

REFERENCES FOR SECTION 15.0

1. DiNunno, J. J., et al., "Calculation for Distance Factors for Power and Test Reactor Sites", TID-14844, March 1962.

2. CE-CES-69, "PDQ-XD", September, 1986.
~~ORNL-4628, "ORIGEN Yields and Cross Sections - Nuclear Transmutation and Decay Data From ENDF/B-IV", Radiation Shielding Information Center, Oak Ridge National Laboratory, September 1975.~~

3. RSIC-DLC-38, "ORIGEN Yields and Cross Sections - Nuclear Transmutation and Decay Data from ENDF/B-IV", Radiation Shielding Information Center, Oak Ridge National Laboratory, September 1975.

4. Chelemer, H., Boman, L. J., Sharp, D. R., "Improved Thermal Design Procedures", WCAP-8567, July, 1975.

5. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors", 10CFR50.46 and Appendix K of 10CFR50. Federal Register, Volume 39, Number 3, January 4, 1974.

6. Bordelon, F. M., et al, "SATAN-VI Program: Comprehensive Space - Time Dependent Analysis of Loss of Coolant", WCAP-8302 (Proprietary) and WCAP-8306 (Non-Proprietary), June 1974.

7. Bordelon, F. M., et al, "LOCTA-IV Program: Loss of Coolant Transient Analysis", WCAP-8301 (Proprietary) and WCAP-8305 (Non-Proprietary), June 1974.

8. Hargrove, H.G., "FACTRAN - A Fortran IV Code for Thermal Transients In A UO_2 Fuel Rod", WCAP-7908, June, 1972.

9. Burnett, T. W. T., McIntyre, C. J., Buker, J. C., Rose, R. P., "LOFTRAN Code Description", WCAP-7907, June, 1972.

10. Risher, D. H., Jr., Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code", WCAP-7979-P-A, WCAP-8028-A, January, 1975.

11. "Westinghouse Nuclear Energy Systems Division Quality Assurance Plan", WCAP-8370-A, August, 1984.

12. S. S. Kilborn, Westinghouse Letter of 1/27/88, DCP-88-508.

Table 15.0.9-1

Iodine and Noble Gas Inventory in Reactor Core and Fuel Rod Gaps*

Nuclide	Core Inventory (Curies)	Fraction of Inventory in Gap** (%)	Gap Inventory (Curies)
I-131	~~1.0E08~~ 8.9E07	.10	~~1.0E07~~ 8.9E06
I-132	~~1.5E08~~ 1.3E08	.10	~~1.5E07~~ 1.3E07
I-133	~~2.1E08~~ 1.9E08	.10	~~2.1E07~~ 1.9E07
I-134	~~2.3E08~~ 2.1E08	.10	~~2.3E07~~ 2.1E07
I-135	~~2.0E08~~ 1.8E08	.10	~~2.0E07~~ 1.8E07
Xe-131m	~~1.4E06~~ 1.2E06	.10	~~1.4E05~~ 1.2E05
Xe-133m	~~6.3E06~~ 5.6E06	.10	~~6.3E05~~ 5.6E05
Xe-133	~~2.1E08~~ 1.9E08	.10	~~2.1E07~~ 1.9E07
Xe-135m	~~3.9E07~~ 3.4E07	.10	~~3.9E06~~ 3.4E06
Xe-135	~~2.2E08~~ 1.9E08	.10	~~2.2E07~~ 1.9E07
Xe-138	~~1.8E08~~ 1.6E08	.10	~~1.8E07~~ 1.6E07
Kr-83m	~~1.3E07~~ [illegible]	.10	~~1.3E06~~ 1.1E06
Kr-85m	~~2.9E07~~ 2[illegible]	.10	~~2.9E06~~ 2.5E06
Kr-85	~~7.2E05~~ 4.3E05	.30	~~2.2E05~~ 1.3E05
Kr-87	~~5.4E07~~ 4.7E07	.10	~~5.4E06~~ 4.7E06
Kr-88	~~7.4E07~~ 6.5E07	.10	~~7.4E06~~ 6.5E06
Kr-89	~~9.4E07~~ 8.2E07	.10	~~9.4E06~~ 8.2E06

* Based on an equilibrum cycle core at end of life. The seven-region core operates at a power level of 3636 MWt and an average cycle burnup of 10,500 MWD/MTU.

** NRC assumption in Regulatory Guide 1.25

Table 15.0.12-1 (Page 1)

Offsite Doses (Rem)

Accident	FSAR Section	Exclusion Area Boundary Whole Body	Exclusion Area Boundary Thyroid	Low Population Zone Whole Body	Low Population Zone Thyroid
Main Steam Line Break	15.1.5				
Case 1 (No iodine spike)		8.6E-2	7.6	4.4E-3	2.6E-1
Case 2 (Pre-spike)		1.03-2	4.22	9.38-4	3.23-1
Case 3 (Coincident spike)		1.26-2	3.32	2.29-3	5.77-1
Loss of Power	15.2.6				
Case 1 (No iodine spike)		4.5E-3	7.0E-2	5.9E-4	6.5E-3
Case 2 (Pre-spike)		4.5E-3	7.3E-2	5.9E-4	7.6E-3
Case 3 (Coincident spike)		4.5E-3	7.2E-2	5.9E-4	8.2E-3
Rod Ejection Accident	15.4.8				
Primary Side Release		~~7.2E-2~~ 2.8E-1	~~5.5~~ 5.9	~~1.1E-2~~ 3.2E-2	~~1.2~~ 6.7E-1
Secondary Side Release		3.3E-2 2.2	~~1.2~~ 1.8E+1	~~1.1E-3~~ 1.6E-1	3.8E-2 6.0
Instrument Line Break	15.6.2				
Case 1 (No iodine spike)		1.6E-1	3.2E-1	5.1E-3	1.0E-2
Case 2 (Pre-spike)		1.8E-1	1.9E+1	6.0E-3	6.3E-1
Case 3 (Coincident spike)		1.8E-1	5.2	6.0E-3	1.7E-1
Steam Generator Tube Rupture	15.6.3				
Case 1 (No iodine spike)		6.4E-1	1.5	2.1E-2	8.8E-2
Case 2 (Pre-spike)		7.1E-1	4.4E+1	2.4E-2	1.5
Case 3 (Coincident spike)		7.0E-1	1.2E+1	2.3E-2	4.6E-1
Loss of Coolant Accident	15.6.5				
Case 1 (With ECCS leakage)		~~5.[illegible]~~ 9.1	~~1.5E+2~~ 1.3E+2	~~9.4E-1~~ 1.1	~~3.4E+1~~ 3.2E+1
Case 2 (Without ECCS leakage)		~~5.3~~ 9.1	~~1.4E+2~~ 1.2E+2	~~9.4E-1~~ 1.1	~~2.9E+1~~ 1.3E+1
Waste Gas Decay Tank Rupture	15.7.1	5.0E-1	-	1.6E-2	-
Locked Rotor	15.3.3				
Case 1 (No Iodine Spike)		4.4E-1	3.6	3.2E-2	1.2
Case 2 (Pre-Spike)		4.4E-1	3.7	3.2E-2	1.2

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Table 15.3.3-2 (Page 1)

Parameters for Postulated Locked Rotor Analysis

1. Data and and assumptions used to estimate radioactive source from postulated accident	Conservative
a. Power Level (MWt)	3565
b. Percent of fuel defected	1
c. Total steam generator tube leak rate during accident and initial 8 hours	1 gpm
d. Activity released to reactor coolant from failed fuel	10% ~~8%~~ of gap inventory
e. Offsite Power	Not available
f. Reactor coolant activity prior to accident	Primary and Secondary Activity During Normal Operations (Table 11.1.1-4)
2. Data and assumptions used to estimate activity released	
a. Iodine partition factor	0.01 ~~0.1~~
b. Initial steam release from 4 steam generators	515,247 lb (0-2 hr) 1,040,910 lb (2-8 hr)
c. Duration of plant cooldown by secondary system after accident, (hrs)	8
3. Dispersion data	
a. Distance to exclusion area boundary (m)	762
b. Distance to low population zone (m)	6096
c. χ/Q at exclusion area boundary (sec/m^3)	5.5E-04
d. χ/Q at low population zone (sec/m^3)	1.8E-05
4. Dose data	
a. Method of dose calculations	Regulatory Guide 1.4
b. Dose conversion assumptions	Regulatory Guides 1.4 & 1.109

Table 15.3.3-2 (Page 2)

Parameters for Postulated Locked Rotor Analysis

	Conservative
c. Doses (Rem)	
Case 1 (No iodine spike)	
Exclusion area boundary	
Whole body	~~4.5E-03~~ 4.4E-01
Thyroid	~~7.0E-02~~ 3.6
Low population zone	
Whole body	~~5.9E-04~~ 3.2E-02
Thyroid	~~6.5E-03~~ 1.2
Case 2 (With pre-existing iodine spike)	
Exclusion area boundary	
Whole body	~~4.5E-03~~ 4.4E-1
Thyroid	~~7.3E-02~~ 3.7
Low population zone	
Whole body	~~5.9E-04~~ 3.2E-2
Thyroid	~~7.6E-03~~ 1.2
~~Case 3 (With coincident iodine spike)~~	
~~Exclusion area boundary~~	
~~Whole body~~	~~4.5E-03~~
~~Thyroid~~	~~7.2E-02~~
~~Low population zone~~	
~~Whole body~~	~~5.9E-04~~
~~Thyroid~~	~~8.2E-03~~

15.4.3.3 Environmental Consequences

The most limiting rod cluster control assembly misoperation, accidental withdrawal of a single RCCA, is predicted to result in less than ~~1%~~ 5% fuel clad damage. The subsequent reactor and turbine trip would result in atmospheric steam dump, assuming the condenser was not available for use. The radiological consequences from this event would be no greater than the ~~main steam line break~~ locked Rotor event, analyzed in Section ~~15.1.5~~ 15.3.3. The February 1983 Catawba SER (page 15-7) concludes that the analysis and calculated consequences for the inadvertent withdrawl of a single RCCA, for which there ~~are~~ is very limited departure from nucleate boiling, is acceptable for a fault of infrequent occurrence.

15.4.3.4 Conclusions

For cases of dropped RCCAs or dropped banks, for which the reactor ~~is tripped~~ by the power range negative neutron flux rate trip, there is no reduction in the margin to core thermal limits, and consequently the DNB design basis is met. It is shown for all cases which do not result in reactor trip that the DNBR remains greater than the limit value and, therefore, the DNB design is met.

For all cases of any RCCA fully inserted, or bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the limit value.

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNB is 5 percent of the total fuel rods in the core.

15.4.4 STARTUP OF AN INACTIVE REACTOR COOLANT PUMP AT AN INCORRECT TEMPERATURE

15.4.4.1 Identification of Causes and Accident Description

If the plant is operating with one pump out of service, there is reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power, and assuming the secondary side of the steam generator in the inactive loop is not isolated, there is a temperature drop across the steam generator in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Administrative procedures require that the unit be brought to a load of less than 25 percent of full power prior to starting the pump in an inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature. Starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which would cause a reactivity insertion and subsequent power increase.

Should the startup of an inactive reactor coolant pump accident occur, the transient will be terminated automatically by a reactor trip on low coolant

2. Beginning of Cycle, Zero Power

Control bank D was assumed to be fully inserted and banks B and C were at their insertion limits. The worst ejected rod is located in control bank D and has a worth of 0.78% Δk/k and a hot channel factor of 11.0. The peak clad average temperature reached 2696°F, the fuel center temperature was 4140°F.

3. End of Cycle, Full Power

Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factors were conservatively calculated to be 0.25% Δk/k and 5.90 respectively. This resulted in a peak clad temperature of 2276°F. The peak hot spot fuel temperature reached melting, conservatively assumed at 4800°F. However, melting was restricted to less than 10% of the pellet.

4. End of Cycle, Zero Power

Control bank D was assumed to be fully inserted and bank C was at its insertion limit. The ejected rod has a worth of 0.90% Δk/k and a hot channel factor of 19.0 respectively. The peak clad and fuel center temperatures were 2586 and 3845°F respectively. The Doppler weighting factor for this case is significantly higher than for the other cases, due to the very large transient hot channel factor.

A summary of the cases presented above is given in Table 15.4.8-1. The nuclear power and hot spot fuel and clad temperature transients for the worst cases are presented in Figures 15.4.8-1 through 15.4.8-4. (Beginning of life full power and beginning of life zero power).

The calculated sequence of events for the worst case rod ejection accidents, as shown in Figures 15.4.8-1 through 15.4.8-4, is presented in Table 15.4.1-1. For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. As discussed previously in Section 15.4.8.2.2, the reactor will remain subcritical following reactor trip.

The ejection of an RCCA constitutes a break in the Reactor Coolant System, located in the reactor pressure vessel head. The effects and consequences of loss of coolant accidents are discussed in Section 15.6.5. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other loss of coolant accident to recover from the event.

Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than ~~10~~ 50 percent of the rods entered DNB ~~based on a detailed three dimensional THINC analysis (Reference 10)~~.

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits (Reference 10). Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the Reactor Coolant System.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are undermoderated, and bowing will tend to increase the undermoderation at the hot spot. Since the 17 x 17 fuel design is also undermoderated, the same effect would be observed. In practice, no significant bowing is anticipated since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling sufficient to distort the lattice is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Environmental Consequences

A conservative analysis for a postulated rod ejection accident is performed to determine the resulting radiological consequences. The analysis is based on a instantaneous fission product release to the reactor coolant of the gap activity from 10 percent of the fuel rods in the core.~~, plus the activity from an assumed 0.25 percent core melt.~~

Prior to the postulated rod ejection accident, it is assumed that the plant is operating at equilibrium levels of radioactivity in the primary and secondary systems with 1 percent fuel defects and a steam generator tube leak rate of 1 gpm. Following the accident, two activity release paths contribute to the total radiological consequences. The first release path is via containment leakage resulting from release of activity from the primary coolant to the containment. The second path is the contribution of contaminated steam in the secondary system dumped through the relief valves, since offsite power is assumed to be lost.

The following conservative assumptions are used in the analysis of the release of radioactivity to the environment in the event of a postulated rod ejection accident. A summary of parameters used in the analysis is given in Table 15.4.8-2.

1. Fifty ~~Ten~~ percent of the gap activity is released. ~~to the containment atmosphere.~~

2. ~~50 percent of the iodines and 100 percent of the noble gases in the melted fuel are released~~. no fuel melting is calculated to occur.

3. ~~50 percent of the iodine released are deposited in the sump~~. The release location for each isotope group is 100 percent noble gases and 25 percent iodines to the containment atmosphere and 100 percent noble gases and 50 percent iodines to the primary coolant.

4. Annulus activity which is exhausted prior to the time at which the annulus reaches a negative pressure of -0.25 in.w.g. is unfiltered.

5. ~~ECCS leakage occurs at twice the maximum operational leakage~~. The iodine chemical species fraction is 0.91 elemental, 0.05 particulate, and 0.04 organic.

~~6. ECCS leakage begins at the earliest possible time sump recirculation can begin.~~

6. ~~7~~. Bypass leakage is 7 percent.

7. ~~8~~. The effective annulus volume is 50 percent of the actual volume.

8. ~~9~~. The annulus filters become fouled at 900 seconds resulting in a 15 percent reduction in flow.

9. ~~10~~. Elemental iodine removal by the ice condenser begins at 600 seconds and continues for 3328.3 seconds with a removal efficiency of 30 percent.

10. ~~11~~. One of the containment air return fans is assumed to fail.

11. ~~12~~. The containment leak rate is 50 percent of the Technical Specifications limit after 1 day.

12. ~~13~~. ~~Iodine partition factor for ECCS leakage is 0.1 for the course of the accident.~~ Iodine removal credit by containment sprays is taken for elemental and particulate iodine.

~~14. No credit is taken for the auxiliary building filters for ECCS leakage.~~

~~15. The redundant hydrogen recombiners and igniters fail. Therefore, purges are required for hydrogen control.~~

(The following assumptions apply to the secondary side analysis).

13. ~~16~~. All the activity released is mixed instantaneously with the entire reactor coolant volume.

14. ~~17~~. The primary to secondary leak rate is 1 gal/min.

15. ~~18~~. The iodine partition factor is ~~0.1~~ 0.01.

16. ~~19~~. ~~The steam release terminates in 120 seconds~~. The duration of plant cooldown by the secondary system is 8 hours.

17. ~~20~~. All noble gases which leak to the secondary side are released.

18. ~~21~~. The primary and secondary coolant concentrations are at the maximum allowed by technical specifications.

Based on the foregoing model, the primary and secondary side releases may be calculated as well as the offsite doses. The doses, given in Table 15.4.8-2,

Table 15.4.8-2 (Page 1)

Parameters for Postulated Rod Ejection Accident Analysis

		Conservative	Realistic
1.	Data and assumptions used to estimate radioactive source from postulated accidents		
a.	Power level (MWt)	3565.	3565.
b.	Percent of fuel defected	1.	0.12
c.	Steam generator tube leak rate prior to and during steam dump (gpm)	1.	0.008
d.	Failed fuel	50 ~~10~~ percent of fuel rods in core	same
e.	Activity released to reactor coolant from failed fuel and available for release		
	Noble gases	50 ~~10~~ percent of core gap inventory	same
	Iodines	25 ~~10~~ percent of core gap inventory	same
f.	Melted fuel	0.0 ~~0.25~~ percent of core	0.
g.	Activity released to reactor coolant from melted fuel and available for release to containment		
	Noble gases	0.0 ~~0.25~~ percent of core inventory	0.
	Iodines	0.0 ~~0.125~~ percent of core inventory	0.
h.	Iodine Fractions (organic, elemental, and particulate)	Regulatory Guide 1.4	same
2.	Data and assumptions used to estimate activity released		
a.	Containment Free volume (ft^3)	1.015E+06	same
b.	Containment leak rate	0.3 percent of containment volume per day, $0 \leq t \leq 24$ hr	0.05 percent of containment volume per day, $0 \leq t \leq 24$ hr

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Table 15.4.8-2 (Page 2)

Parameters for Postulated Rod Ejection Accident Analysis

		Conservative	Realistic
		0.15 percent of containment volume per day, t>24 hr	0.025 percent of containment volume per day, t>24 hr
c.	Bypass leakage fraction	0.07	0.07
d.	Iodine partition factor for steam release	~~0.1~~ 0.01	-
e.	Offsite power	Lost	-
f.	Plant cooldown by secondary (hrs.) ~~Steam dump from relief valves (lb)~~	~~44500.~~ 8.	-
~~g.~~	~~Duration of dump from relief valves (sec)~~	~~120.~~	-
3.	Dispersion data		
a.	Distance to exclusion area boundary (m)	762.	762.
b.	Distance to low population zone (m)	6096.	6096.
c.	x/Q at exclusion area boundary (sec/m³)		
	0-2 hrs	5.5E-04	1.3E-04
d.	x/Q at low population zone (sec/m³)		
	0-8 hrs	1.8E-05	6.2E-06
	8-24 hrs	1.2E-05	5.4E-06
	1-4 days	4.3E-06	2.5E-06
	4+days	1.2E-06	9.7E-07
4.	Dose data		
a.	Method of dose calculation	Regulatory Guide 1.77	same
b.	Dose conversion assumptions	Regulatory Guides 1.4 and 1.109	same
c.	Doses (Rem)		
	Primary side		
	Exclusive area boundary		
	Whole body	~~7.2E-2~~ 2.8E-1	
	Thyroid	~~5.5~~ 5.9	

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Table 15.4.8-2 (Page 3)

Parameters for Postulated Rod Ejection Accident Analysis

	Conservative	Realistic
Low population zone		
Whole body	~~1.1E-02~~ 3.2E-2	
Thyroid	~~1.2~~ 6.7E-1	
Secondary side		
Exclusion area boundary		
Whole body	~~3.3E-02~~ 2.2	
Thyroid	~~1.2~~ 1.8E+1	
Low population zone		
Whole body	~~1.1E-03~~ 1.6E-1	
Thyroid	~~3.8E-02~~ 6.0	

Offsite Dose Consequences

The offsite radiological consequences of a LOCA are calculated based on the following assumptions and parameters:

1. 100 percent of the core noble gases and 25 percent of the core iodines are released to the containment atmosphere.

2. 50 percent of the core iodines are deposited in the sump.

3. The iodine chemical species fraction is 0.91 elemental, 0.05 particulate, and 0.04 organic.

4. ~~3~~. Annulus activity which is exhausted prior to the time at which the annulus reaches a negative pressure of -0.25 in. w.g. is unfiltered.

5. ~~4~~. ECCS leakage begins at the earliest possible time sump recirculation can begin.

6. ~~5~~. ECCS leakage occurs at twice the maximum operational leakage. Also included is leakage from a gross failure of a passive component. The leakage is conservatively assumed to be 50 gallons per minute, starting at 24 hours after the LOCA and lasting for 30 minutes.

7. ~~6~~. Bypass leakage is 7 percent.

8. ~~7~~. The effective annulus volume is 50 percent of the actual volume.

9. ~~8~~. The annulus filters become fouled at 900 seconds resulting in a 15 percent reduction in flow.

10. ~~9~~. Elemental iodine removal by the ice condenser begins at 600 seconds and continues for ~~3328.9~~ 4191 seconds with a removal efficiency of 30 percent.

11. ~~10~~. One of the containment air return fans is assumed to fail.

12. ~~11~~. The containment leak rate is fifty percent of the Technical Specification limit after 1 day.

13. ~~12~~. Iodine partition factor for ECCS leakage is 0.1 for the course of the accident.

14. ~~13~~. No credit is taken for the auxiliary building filters for ECCS leakage.

15. ~~14~~. ~~The redundant hyrdogen recombiners and igniters fail. Therefore, purges are required for hydrogen control.~~ Iodine removal credit by containment sprays is taken for elemental and particulate iodine.

The doses are presented in Table 15.6.5-9 and are within the ~~limits of~~ guideline values of 10 CFR 100.

Control Room Operator Dose

The maximum postulated dose to a control room operator is determined based on the containment releases of a Design Basis Accident. In addition to the parameters and assumptions listed above, the following apply:

1. The control room pressurization rate is ~~4,000~~ 2,800 cfm; the filtered recirculation rate is 2,000 cfm. 1,200 cfm serves the Control room area (cable and electrical penetration rooms).
2. The unfiltered inleakage into the control room is 10 cfm.
3. Other assumptions are listed in Table 15.6.5-10.

15.6.6 A NUMBER OF BWR TRANSIENTS

Not applicable to Catawba.

Table 15.6.5-9 (Page 1)

Parameters for LOCA Offsite Dose Analysis

	Conservative	Realistic
1. Data and assumptions used to estimate radioactive source from postulated accidents		
a. Power level	3565.	3565.
b. Fail fuel rods in core	100% of fuel rods in core	2% of fuel
c. Activity released to reactor coolant from failed fuel and available for release		
Noble gases activity	100% of core activity	2% of core
Iodines activity	50% of core activity	2% of core
d. Iodine fractions (organic, elemental, and particulate)	Regulatory Guide 1.4	same
2. Data and assumptions used to estimate activity released		
a. Containment free volume		
Upper containment volume (ft^3)	6.70+05	same
Lower containment volume (ft^3)	3.45E+06	same
Total containment free volume (ft^3)	1.015E+06	same
b. Iodine activity released to containment	25%	same
c. Containment leak rate	0.3% of containment volume per day, $0 \leq t \leq 24$ hr	0.05% of containment volume per day, $0 \leq t \leq 24$ hr
	0.15% of containment volume per day, t>24 hr	0.025% of containment volume per day, t>24 hr
d. Bypass leakage fraction	0.07	0.07
e. Annulus ventilation iodine filter efficiency	95%	99%

Table 15.6.5-9 (Page 3)

Parameters for LOCA Offsite Dose Analysis

	Conservative	Realistic
c. Doses (Rem)		
Case 1 (with ECCS leakage)		
Exclusion Area Boundary		
Whole Body	~~5.4~~ 9.1	
Thyroid	~~1.5E+2~~ 1.3E+2	
Low Population Zone		
Whole Body	~~9.4E-1~~ 1.1	
Thyroid	~~3.4E+1~~ 3.2E+1	
Case 2 (Without ECCS leakage)		
Exclusion Area Boundary		
Whole Body	~~5.3~~ 9.1	
Thyroid	~~1.4E+2~~ 1.2E+2	
Low Population Zone		
Whole Body	~~9.4E-1~~ 1.1	
Thyroid	~~2.9E+1~~ 1.3E+1	

Table 15.6.5-10 (Page 2)

Parameters for LOCA Control Room Dose Analysis

		Conservative	Realistic
3.	Dispersion data - Control room intake χ/Q (sec/m^3)		
	0-8 hrs	9.9E-04	
	8-24 hrs	7.2E-04	
	1-4 days	5.1E-04	
	4+ days	2.8E-04	
4.	Dose data		
	a. Method of dose calculations	Standard Review Plan 6.4	
	b. Dose conversion assumptions	Regulatory Guides 1.4, 1.109	
	c. Doses (Rem)		
	Whole body	~~8.5E-1~~ 1.6	
	Thyroid	~~2.0E+1~~ 1.4E+1	
	Skin	~~1.7E+1~~ 3.2E+1	

9. STARTUP PHYSICS TESTING

The standard scope of reload startup physics testing conducted at the Catawba Unit 1 is summarized below:

Zero Power Physics Testing (ZPPT)

- All Rods Out Critical Boron Concentration (AROCBC)
- Isothermal Temperature Coefficient (ITC)
- Control Rod Bank Worth
- Differential Boron Worth (DBW)

Power Escalation Testing (PET)

- Flux Symmetry Check (Low Power, eg. 30 %FP)
- Core Power Distribution -- CPD (Intermediate Power)
- CPD (Full Power)
- AROCBC (Full Power)

BWFC has reviewed the startup physics testing program (scope, test methods, and acceptance criteria) for Catawba. All aspects of the existing program are acceptable with respect to implementation oı the BWFC licensing analyses and a complete reload batch of Mark-BW fuel assemblies. Therefore, operation with either a mixed Westinghouse and BWFC core or future cores with all BWFC fuel will not require any changes to the current Duke startup physics testing program.

10.0 REFERENCES

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