RELOAD REPORT

Catawba Unit 2 Cycle 7

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the seventh cycle of Catawba Nuclear Station, Unit 2 at the rated core power level of 3411 MW_{th}. Included are the required analyses as outlined in the USNRC document "Guidance for Proposed License Amendments Relating to Refueling," July 1975.

The incoming Mark-BW fuel for Cycle 7 is the second Catawba Unit 2 reload batch supplied by B&W Fuel Company (BWFC). To support implementation of Mark-BW fuel in the McGuire and Catawba nuclear stations, Duke Power Company (DPC) developed methods and models that are used to analyze the plants during normal and off-normal operation. The thermal-hydraulic analytical models are documented in topical report DPC-NE-3000P (Reference 11) and DPC-NE-3002-A (Reference 16) for non-LOCA transients and BAW-10174-A (Reference 13) for LOCA. Portions of the analytical methodology are documented in topical reports DPC-NE-3001-PA (Reference 12) and DPC-NE-2004-PA (Reference 8).

Section 2 of this report describes the operating history for fuel in Catawba Unit 2. 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 and Core Operating Limits Report (COLR) are provided in Section 8. The scope of Startup Physics Testing for Catawba Unit 2, Cycle 7 is provided in Section 9.

All of the accidents analyzed in the Final Safety Analysis Report FSAR (Reference 1) have been reviewed and are applicable for Cycle 7 operation. In those cases where Cycle 7 characteristics were conservative compared to those analyzed for previous cycles, new analyses were not performed. With the exception of the post-LOCA subcriticality and dropped rod analyses, the Cycle 7 thermal-hydraulic and physics parameters are bounded by the existing Catawba FSAR Chapter 15 analyses. The results of reanalyzed accidents for Catawba Unit 2 Cycle 7 are discussed in Section 7.

Amendment Number 74 (Unit 1) and Amendment Number 68 (Unit 2) to the Catawba Nuclear Station Facility Operating License 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 core operating limits are made via the Core Operating Limits Report.

The Technical Specifications have been reviewed, and the modifications required for Cycle 7 are given in Section 8. Based on the analyses performed, it has been concluded that Catawba Unit 2 Cycle 7 can be safely operated at a core power level of 3411 MW_{th}.

2. OPERATING HISTORY

The current operating cycle for Catawba Unit 2 is Cycle 6, which achieved criticality on March 30, 1993 and reached 100% full power on April 5, 1993. Cycle 6 is scheduled to shut down in April 1994 after 380 EFPD. No operating anomalies have occurred during Cycle 6 operations that would adversely affect fuel performance in Cycle 7.

Catawba Unit 2 Cycle 7 is scheduled to start up in June 1994 at a rated power level of 3411 MW_{th} and has a design cycle length of 430 ± 10EFPD.

3. GENERAL DESCRIPTION

The Catawba Unit 2 reactor core is described in detail in Chapter 4 of the FSAR (Reference 1). The core consists of 193 assemblies, each of which is a 17X17 array containing 264 fuel rods, 24 guide tubes, and 1 incore instrument tube. The Catawba 2 Cycle 7 core has 105 burned assemblies and 88 fresh assemblies. The fuel rod outside diameters are 0.360 and 0.374 inch, and the clad thicknesses are 0.0225 and 0.024 inch for the Westinghouse optimized fuel assembly (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 design loadings are 423.5 and 456.3 kg of uranium per assembly for OFA and Mark-BW fuel, respectively. The initial design enrichments of batches 7A and 8A were both 3.75 w/o ^{235}U. The design enrichment of the fresh batch 9A (Mark-BW) includes 40 assemblies with 3.50, 40 assemblies with 4.00 and 0.71 axial blankets and 8 assemblies with 3.60 and 0.71 axial blankets.

Figure 3-1 gives the full core loading pattern for Cycle 7. The 29 batch 7A and 76 batch 8A assemblies will be shuffled to new locations. The 88 fresh batch 9A assemblies will be loaded into the core in a symmetric checkerboard pattern. Figure 3-2 is a quarter core map showing the burnup and region number of each assembly at the beginning of Cycle 7. Figure 3-2 also provides batch average enrichment and burnup.

Cycle 7 will be operated in a feed-and-bleed mode. Core reactivity is controlled by 53 rod cluster control assemblies (RCCAs), 1536 Mark-BW burnable absorbers, and soluble boron shim. The Cycle 7 locations of the 53 rod cluster control assemblies with their respective designations are unchanged from the previous cycle. The Cycle 7 locations of Mark-BW BPRA clusters and number of pins enriched to 3.0 w/0, 2.5 w/0, and 2.0 w/0 B_4C-Al_2O_3 are also shown in Figure 3-3.

FIGURE 3-1
CORE LOADING PATTERN FOR CATAWBA UNIT 2 CYCLE 7

PREVIOUS CORE LOCATIONS
REGION NUMBERS

Z-ZZ YY	CYCLE 6 LOCATION REGION NUMBER

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					K-05 8	F 9	A-07 7	F 9	R-07 7	F 9	F-05 8				
2			C-14 7	E-14 8	F 9	J-14 8	F 9	F-01 8	F 9	G-14 8	F 9	L-14 8	N-14 7		
3		B-13 7	A-06 8	F 9	G-04 8	F 9	M-03 8	F 9	D-03 8	F 9	J-04 8	F 9	F-15 8	P-13 7	
4		B-11 8	F 9	F-13 8	F 9	G-06 8	F 9	R-11 7	F 9	J-06 8	F 9	N-10 8	F 9	P-11 8	
5	L-06 8	F 9	M-09 8	F 9	R-08 7	F 9	M-11 8	F 9	D-11 8	F 9	H-01 7	F 9	D-09 8	F 9	E-06 8
6	F 9	B-07 8	F 9	K-09 8	F 9	C-10 8	F 9	H-07 8	F 9	K-13 8	F 9	F-09 8	F 9	P-07 8	F 9
7	J-15 7	F 9	N-04 8	F 9	E-04 8	F 9	R-05 7	F 9	E-01 7	F 9	L-04 8	F 9	C-04 8	F 9	G-15 7
8	F 9	R-06 8	F 9	E-15 7	F 9	J-08 8	F 9	H-11 7	F 9	G-08 8	F 9	L-01 7	F 9	A-10 8	F 9
9	J-01 7	F 9	N-12 8	F 9	E-12 8	F 9	L-15 7	F 9	A-11 7	F 9	L-12 8	F 9	C-12 8	F 9	G-01 7
10	F 9	B-09 8	F 9	K-07 8	F 9	F-03 8	F 9	H-09 8	F 9	N-06 8	F 9	F-07 8	F 9	P-09 8	F 9
11	L-10 8	F 9	M-07 8	F 9	H-15 7	F 9	M-05 8	F 9	D-05 8	F 9	A-08 7	F 9	D-07 8	F 9	E-10 8
12		B-05 8	F 9	C-06 8	F 9	G-10 8	F 9	A-05 7	F 9	J-10 8	F 9	K-03 8	F 9	P-05 8	
13		B-03 7	K-01 8	F 9	G-12 8	F 9	M-13 8	F 9	D-13 8	F 9	J-12 8	F 9	R-10 8	P-03 7	
14			C-02 7	E-02 8	F 9	J-02 8	F 9	K-15 8	F 9	G-02 8	F 9	L-02 8	N-02 7		
15					K-11 8	F 9	A-09 7	F 9	R-09 7	F 9	F-11 8				

F without row designator indicates fresh fuel assembly

FIGURE 3-2

ENRICHMENT AND BOC BURNUP DISTRIBUTION FOR CATAWBA 2 CYCLE 7

	H	G	F	E	D	C	B	A
08	7 30.0846 34.1401	9 .0000 .0000	8 18.8930 19.5371	9 .0000 .0000	7 27.9524 33.2969	9 .0000 .0000	8 11.6970 15.9628	9 .0000 .0000
09	9 .0000 .0000	7 27.9327 33.2861	9 .0000 .0000	8 18.8283 19.8595	9 .0000 .0000	8 17.1522 19.2534	9 .0000 .0000	7 30.3127 33.9631
10	8 18.8138 19.4642	9 .0000 .0000	8 19.2502 19.9436	9 .0000 .0000	8 18.9375 19.7501	9 .0000 .0000	8 17.4605 19.3905	9 .0000 .0000
11	9 .0000 .0000	8 18.8812 19.9084	9 .0000 .0000	7 23.2069 24.9092	9 .0000 .0000	8 18.8532 19.9977	9 .0000 .0000	8 19.1222 19.8003
12	7 27.9337 33.2875	9 .0000 .0000	8 18.8914 19.7078	9 .0000 .0000	8 19.2604 19.9516	9 .0000 .0000	8 16.2175 19.5958	
13	9 .0000 .0000	8 17.1834 19.3189	9 .0000 .0000	8 18.8119 19.9596	9 .0000 .0000	8 11.6963 15.9598	7 26.4220 31.3529	
14	8 11.6891 15.9514	9 .0000 .0000	8 17.4501 19.3704	9 .0000 .0000	8 16.2252 19.6075	7 26.4991 31.4341	FUEL BATCH LABEL AVG ASSY EXPOSURE MAX PIN EXPOSURE	
15	9 .0000 .0000	7 30.2625 33.9490	9 .0000 .0000	8 19.1730 19.9036				

Region	Enrichment w/o U-235	Cycles Burned	Number of Assemblies	BOC Burnup GWD/MTU
7A	3.75(OFA)	2	29	27.600
8A	3.75	1	76	17.576
9A	4.00/0.71*	0	40	0
9B	3.60/0.71*	0	8	0
9C	3.50	0	40	0
Core	N/A	N/A	193	11.068

*These are Nat'l U blanketed fuel assemblies.

FIGURE 3-3
CATAWBA UNIT 2 CYCLE 7
BURNABLE ABSORBER AN^ SOURCE ASSEMBLY LOCATIONS

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1						0		0		0					
2					8+		16+		16+		8+				
3				12+		24+		24&		24+		12+			
4			12+		24*		24*		24*		24*		12+		
5		8+		24*		24*		24*		24*		24*		8+	SS
6	0		24+		24*		24*		24*		24*		24+		0
7		16+		24*		24*		24*		24*		24*		16+	
8	0		24&		24*		24*		24*		24*		24&		0
9		16+		24*		24*		24*		24*		24*		16+	
10	0		24+		24*		24*		24*		24*		24+		0
11	SS	8+		24*		24*		24*		24*		24*		8+	
12			12+		24*		24*		24*		24*		12+		
13				12+		24+		24&		24+		12+			
14					8+		16+		16+		8+				
15						0		0		0					

Number of Burnable Absorber Pins per Assembly	BP enrichment (B4C w/o)	Number of Backplates
8	2.00	8
12	2.00	8
16	2.00	8
24	2.00	8
24	2.50	4
24	3.00	40
TOTAL = 1536 pins		TOTAL = 76 backplates

SS = Secondary Source
+ = 2.0 w/o BP's
& = 2.5 w/o BP's
* = 3.0 w/o BP's

4. FUEL SYSTEM DESIGN

4.1 Fuel Assembly Mechanical Design

The Catawba 2 Cycle 7 core will include 88 fresh Mark-BW fuel assemblies. A total of forty eight (48) of these fresh assemblies will be natural uranium blanketed. Forty (40) of the blanketed assemblies will have an enrichment of 4.00 wt % U_{235} in the non-blanket region while the remaining eight blanketed assemblies will employ an enrichment of 3.60 wt % U_{235}. Forty (40) fresh fuel assemblies will be non-blanketed and employ an enrichment of 3.50 wt % U_{235}. The re-inserted fuel assemblies in Cycle 7 will be Westinghouse Optimized fuel assemblies (29) and Mark-BW fuel assemblies (76).

The Mark-BW 17 x 17 Zircaloy spacer grid fuel assembly is similar in design to the Westinghouse standard fuel assembly, Reference 2. The fuel rod outer diameter and guide tube top section, dashpot diameters, and instrument tube diameter are the same as the Westinghouse standard 17 x 17 design. The unique features of the Mark-BW design include the Zircaloy intermediate spacer grids, the spacer grid restraint system, and the use of Zircaloy grids with the standard lattice design. Mark-BW fuel design dimensions and parameters for Catawba 2 Cycle 7 are listed in Table 4-1.

4.2 Fuel Rod Design

Duke Power Company has performed generic Mark-BW mechanical analyses using the approved methodologies described in Reference 3. The generic analyses envelope the Catawba 2 Cycle 7 reinsert fuel. Critical Cycle 7 fresh fuel as-built parameters will be compared against values assumed in the generic analyses prior to cycle startup. This will determine the applicability of the analyses to the fresh fuel. The cladding collapse and minimum LHRTM limits in Table 4-1 are based upon these generic analyses.

4.2.1 Fuel Rod Cladding Collapse

The fuel rods were analyzed for creep collapse using the CROV computer code, Reference 4, and the methodology described in Reference 3. Internal pin pressures and clad temperatures used in CROV were calculated using the TACO2 computer code, Reference 5. A conservative power history which envelopes the predicted peaking for the Catawba 2 Cycle 7 fuel was analyzed. The collapse time was conservatively determined to be greater than the maximum predicted residence time for the Mark-BW fuel (Table 4-1).

4.2.2 Fuel Rod Cladding Stress

As described in Reference 3, Duke Power Company has performed a conservative generic fuel rod cladding stress analysis using the ASME pressure vessel stress intensity limits as guidelines. The maximum cladding stress intensities were shown to be within the ASME limits under all loading conditions. The generic Mark-BW cladding stress analysis includes the following conservatisms:

* Conservative cladding dimensions.
* High external pressure.
* Low internal pin pressure.
* High radial temperature gradient through the clad.

4.2.3 Fuel Rod Cladding Strain

Diametral cladding strain resulting from a local power transient is limited to 1.0 %. A generic cladding strain analysis was performed using TACO2 to determine the maximum allowable local power change that the fuel could experience without exceeding the 1.0 % limit. The maximum calculated local power change resulting from a worst case core maneuvering scenario was compared with the maximum allowable power change. This comparison demonstrated that margin exists to the 1% strain limit.

4.3 Thermal Design

The thermal performance of the Mark-BW fuel assemblies was evaluated using TACO2 with the methodology given in Reference 3. The nominal fuel parameters used to determine the generic linear heat rate to centerline melt (LHRTM) limits are given in Table 4-1. The LHRTM analysis included the following bounding conservatisms:

* Maximum gap based on as-fabricated pellet and clad data.

* Maximum incore densification based on resinter test results.

The maximum predicted Mark-BW assembly burnup at EOC 7 (in Batch 8) is 39,688 MWD/MTU and the maximum predicted fuel rod burnup (in Batch 8) is 41,008 MWD/MTU. The fuel rod internal pressure has been evaluated for the highest burnup rod using TACO2 and a conservative pin power history. The maximum internal pin pressure is less than the core exit pressure of 2280 psia.

4.4 Material Design

The fresh Mark-BW fuel is not unique in concept, nor does it utilize different component materials. Thus, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the fresh fuel is identical to that of existing Westinghouse OFA and Mark-BW fuel types.

4.5 Operating Experience

Experience with the Mark-BW 17 x 17 fuel assembly design started with the irradiation of four lead assemblies in McGuire 1 Cycle 5. The assemblies from this program have completed irradiation with a maximum assembly burnup of 42,756 MWD/MTU. The lead assemblies were examined after each operational cycle and the fuel assembly bow, twist, growth, and holddown spring set were all within nominal bounds.

Catawba 2 Cycle 7 will be the ninth reload batch of Mark-BW fuel supplied to Duke Power Company.

Table 4-1. Fuel Design Parameters and Dimensions

Mark-BW

	Batch 8	Batch 9
Nominal fuel rod OD, in.	0.374	0.374
Nominal fuel rod ID, in.	0.326	0.326
Nominal active fuel length, in.	144.0	144.0
Nominal fuel pellet OD, in.	0.3195	0.3195
Fuel pellet initial density, % TD	96.0	96.0
Initial fuel enrichment, wt. % U_{235}	3.75	4.00/0.71* 3.60/0.71* 3.50
Maximum estimated fuel assembly average burnup, (MWd/mtU)	39,688	21,984
Cladding collapse burnup, (MWd/mtU)	>57,748	>57,748
Nominal linear heat rate (LHR), kW/ft	5.43	5.43
Ave. fuel temperature @ nom. LHR, deg F	1360	1360
Minimum LHR to melt, kW/ft:		
0-1000 MWD/MTU	21.5	21.5
> 1000 MWD/MTU	21.8	21.8

*Natural uranium blanketed fuel assemblies (see Section 4.1)

5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 provides the core physics parameters for Cycles 6 and 7. The values were generated using the methodology described in DPC-NE-1004A (Reference 6) and DPC-NE-3001-PA (Reference 12). Cycle 7 values are valid for the design cycle length (430 EFPD ± 10 EFPD). Figure 5-1 illustrates a representative relative power distribution for the beginning of Cycle 7 at full power. This case was calculated as part of the design depletion using the SIMULATE-3P methodology as described in DPC-NE-1004A (Reference 6). This case contained equilibrium xenon and rods in the all rods out (ARO) position.

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 is demonstrated in Table 5-2. The shutdown margin calculations include a 10% uncertainty in the available all rods in (ARI) position minus the most reactive stuck rod worth at HZP. The shutdown calculation at the end of Cycle 7 was analyzed at 440 EFPD (430 EFPD + 10 EFPD window).

5.2. Nuclear Design Methodology

The Cycle 7 physics parameters appearing in this report were calculated with the CASMO-3 and SIMULATE-3P codes. These codes and methods were approved by the NRC as documented in Reference 6. The SIMULATE-3P calculations were performed in three dimensions. The Reactor Protection System (RPS) limits and operational limits for the core were verified by analyses for this fuel cycle using methodology approved by the NRC in Reference 7. The operational limits are provided in the COLR.

Table 5.1 Physics Parameters[a] Catawba 2 Cycles 6 and 7

	Cycle 6	Cycle 7
Design cycle length, EFPD	380	430
Design cycle burnup, MWD/MTU	15390	16788
Design average core burnup - EOC, MWD/MTU	28831	27856
Design initial core loading, MTU	84.2207	87.0983
Critical boron - BOC, ppmb, no Xe[b]		
HZP, ARO	1713	1782
HFP, ARO	1569	1551
Critical boron - EOC, ppmb		
HZP, No Xe, ARO	525	670
HFP, Eq Xe, ARO	0	0
Total Control Rod Worths - HZP, eq Xe pcm		
BOC	6996	6853
EOC[c]	7495	6831
Max ejected rod worth(d) - HZP, pcm		
BOC (D12)	373	472
EOC[c] (F10)	552	383
Max stuck rod worth - HZP, eq Xe pcm		
BOC (F10)	1194	1001
EOC[c] (F10)	1202	1114
Power deficit - HZP to HFP, eq Xe pcm		
BOC	-1690	-1677
EOC[c]	-3018	-2739
Doppler coeff - HFP, pcm/°F		
BOC, no Xe	-1.16	-1.51
EOC[c], eq Xe	-1.45	-1.71
Moderator coeff - HFP, pcm/°F		
BOC, no Xe	-2.93	-6.95
EOC[c], eq Xe, 0 PPMB	-32.50	-35.56
Boron worth - HFP, pcm/ppmb		
BOC	-7.89	-6.89
EOC[c]	-9.22	-8.33

Table 5.1 Physics Parameters[(a)] Catawba 2 Cycles 6 and 7 (cont)

	Cycle 6	Cycle 7
Equilibrium Xenon worth - HFP, pcm		
BOC (4 EFPD)	2604	2641
EOC	2990	2880
Effective delayed neutron fraction - HFP		
BOC	0.006090	0.006369
EOC	>0.00440	0.005248

(a) Cycle 6 and 7 values obtained from Duke Power Company analyses.

(b) HZP denotes hot zero power (core average 557°F Tavg); HFP denotes hot full power (590.8°F vessel Tavg).

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

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

Table 5-2. Shutdown Margin Calculation for Catawba 2 Cycle 7

Control Rod Worth	BOC (PCM)	EOC[a] (PCM)
1. All rods inserted (ARI), HZP	6853	6831
2. ARI less most reactive stuck rod, HZP	5852	5717
3. Less 10% uncertainty	5267	5145
Required Rod Worth		
4. Rod insertion allowance (RIA)	213[b]	316[b]
5. Power defect, HFP to HZP	1934[b]	2998
6. Shutdown margin (total available worth minus total required worth)	3120	1831

NOTE: Required shutdown margin is 1300 PCM.

(a) EOC physics parameters calculated at 440 EFPD, i.e., design EOC plus 10 EFPD.

(b) The rod insertion allowance and power defect include penalties to account for the effects of transient xenon conditions.

Figure 5-1: BOC (4 EFPD), Cycle 7 Two-Dimensional Relative Power Distribution - HFP, Equilibrium Xenon

	H	G	F	E	D	C	B	A
	.8701	1.2230	1.2200	1.3304	1.1157	1.2494	1.2449	1.0887
08	.8193	1.0113	1.1224	1.1658	1.0259	1.1251	1.1630	.7856
	1.0620	1.2093	1.0870	1.1412	1.0875	1.1105	1.0704	1.3858
	M-03	Q-17	N-05	Q-01	E-04	A-01	D-13	B-09
	1.2217	1.0560	1.3330	1.2489	1.3512	1.2860	1.2787	.8278
09	1.0066	.9392	1.1418	1.1844	1.1724	1.1939	1.1066	.5422
	1.2137	1.1244	1.1675	1.0544	1.1525	1.0771	1.1555	1.5268
	A-17	N-13	Q-17	M-15	Q-17	E-14	B-09	A-01
	1.2199	1.3348	1.2748	1.3664	1.2767	1.2831	1.2017	.9727
10	1.1221	1.1412	1.1802	1.1962	1.2013	1.1677	1.0474	.7027
	1.0872	1.1696	1.0802	1.1423	1.0628	1.0988	1.1473	1.3842
	M-14	Q-17	M-14	Q-17	N-13	A-01	C-05	B-06
	1.3305	1.2544	1.3703	1.1527	1.3659	1.2435	1.2363	.8208
11	1.1657	1.1888	1.1999	1.1154	1.1948	1.1595	1.0135	.4473
	1.1414	1.0551	1.1421	1.0335	1.1432	1.0724	1.2199	1.8351
	Q-17	O-13	Q-17	D-13	A-01	C-05	C-07	A-01
	1.1162	1.3601	1.2837	1.3704	1.2776	1.2952	1.0910	
12	1.0265	1.1839	1.2082	1.1983	1.1909	1.1359	.6832	
	1.0874	1.1488	1.0625	1.1436	1.0729	1.1402	1.5969	
	N-05	Q-17	M-14	A-01	D-13	C-04	A-01	
	1.2500	1.2934	1.2911	1.2496	1.3008	1.2196	.6768	
13	1.1259	1.1999	1.1734	1.1657	1.1455	.8576	.3372	
	1.1103	1.0779	1.1003	1.0720	1.1356	1.4220	2.0071	
	Q-01	N-05	A-01	E-03	D-03	A-01	A-01	
	1.2461	1.2821	1.2064	1.2427	1.0986	.6953	MAX PIN POWER	
14	1.1643	1.1085	1.0516	1.0189	.6905	.3428	AVG ASSY POWER	
	1.0703	1.1566	1.1472	1.2197	1.5910	2.0286	PK PIN/ASSY FACTOR	
	E-04	I-02	E-03	G-03	A-01	A-01	MAX PIN LOCATION	
	1.0900	.8277	.9759	.8250				
15	.7867	.5635	.7051	.4491				
	1.3856	1.5230	1.3841	1.8369				
	I-02	A-01	F-02	A-01				

The maximum pin power is 1.3704 in assembly E-12 at pin A-01.

The maximum assembly power is 1.2082 in assembly F-12.

The maximum peak pin to assembly factor is 2.0286 in assembly C-14.

6. THERMAL-HYDRAULIC DESIGN

The generic and cycle-specific analyses supporting Cycle 7 operation were performed by Duke Power Company using the methodology described in Reference 8. Cycle 7 was analyzed using Duke's Statistical Core Design (SCD) methodology. Uncertainties on parameters that affect DNB performance are statistically combined to determine a Statistical DNBR limit (SDL). Using the BWCMV correlation, Reference 9, a generic SDL of 1.40 was calculated using a set of generic uncertainties given in Reference 8. The system parameter uncertainties used in Reference 8 and given in Table 6-1 bound the uncertainties specifically calculated for Catawba.

Reactor core safety limits for Cycle 7 are based on a full Mark-BW core and a design $F\Delta H$ of 1.50. The Cycle 7 nominal thermal-hydraulic design conditions are given in Table 6-2. The C2C7 core will have 48 assemblies with axial blanket fuel. Maximum Allowable Peaking (MAP) limits that specifically address the axial blanket fuel were used in the C2C7 maneuvering analysis.

The Mark-BW fuel assembly was designed to be hydraulically compatible with Westinghouse optimized fuel (OFA). BWFC has performed a series of flow tests to verify the compatibility of the two designs. The tests showed that the total pressure drop across the OFA fuel is 2.4% higher than the pressure drop across the Mark-BW fuel, Reference 2. A generic transition core analysis was performed to determine the DNBR impact of this difference.

Since the Mark-BW fuel has a lower overall pressure drop than the OFA design, a Mark-BW assembly in a mixed core will tend to have more flow through it and consequently more DNB margin than the same assembly in an all Mark-BW core. Conversely, flow will be forced out of the OFA fuel in a mixed core; thus, the need to calculate a DNBR penalty for the OFA fuel. A generic transition core DNBR penalty was determined by modeling a conservative core configuration with one OFA assembly as the hot assembly. The rest of the core was modeled as Mark-BW fuel. A number of statepoints and peaking conditions were analyzed, yielding a maximum DNBR penalty of 3.8% for the OFA fuel.

To provide design flexibility, margin is added to the SDL to determine a design DNBR limit (DDL). For the generic Mark-BW and Catawba 2 Cycle 7 analyses, the DDL is 1.55 (10.7% margin above the SDL). The DNBR penalties, such as the OFA transition core penalty, that must be assessed against the margin are given in Table 6-3.

Prior to C2C7, a DNBR penalty was applied against the margin in the DDL to account for the flow distribution effects of the grid restraint system used for Mk-BW fuel assemblies. This penalty (2.8 %) was conservatively estimated using VIPRE-01. BWFC has now performed several CHF tests which show that a DNBR penalty is not required for the system used to hold the intermediate spacer grids in place.

Table 6-1
System Uncertainties Included in the
Statistical Core Design Analysis

Reference 8

Parameter	Uncertainty	Distribution
Core power	+/- 2 %	Normal
RCS flow	+/- 2.2 %	Normal
Core bypass flow	+/- 1.5 %	Uniform
Pressure	+/- 30 psi	Uniform
Inlet temperature	+/- 4 °F	Uniform

Table 6-2.
Nominal Thermal-Hydraulic Design Conditions

Catawba 2 Cycle 7

Core power, MW_{th}	3411
Core exit pressure, psia	2280
Vessel ave. temperature, °F	590.8
RCS flow, gpm	382,000
Core bypass flow, %	7.5
Reference design $F\Delta H$	1.50
Reference design axial shape	1.55 Cosine
CHF correlation	BWCMV
Statistical DNBR limit	1.40
Design DNBR limit	1.55

Table 6-3.
DNBR Penalties

Statistical DNBR limit	1.40
Design DNBR limit	1.55
DNBR margin	10.7 %

DNBR Penalty	Mark-BW	OFA
Transition core	0 %	3.8 %
Instrumentation	2.8 %	2.8 %
Flow Anomaly	0.5 %	0.5 %
Rod bow	0 %	3.5 %
Total DNBR penalty	3.3 %	10.6 %
Available DNBR Margin	**7.4 %**	**0.1 %**

7. ACCIDENT ANALYSIS

Safety Analysis

Each FSAR accident listed below has been examined with respect to changes in Cycle 7 parameters to determine the effect of the Cycle 7 reload and to ensure that thermal performance during hypothetical transients is not degraded.

- Increase in feedwater flow
- Excessive load increases
- Steam system piping failure
- Turbine trip
- Feedwater system pipe break
- Partial loss of forced reactor coolant flow
- Complete loss of forced reactor coolant flow
- Reactor coolant pump shaft seizure (locked rotor)
- Uncontrolled rod bank withdrawal from subcritical or low power startup condition
- Uncontrolled rod bank withdrawal at power
- Dropped rod/rod bank
- Statically misaligned rod
- Single rod withdrawal
- Startup of an inactive reactor coolant pump
- Boron dilution
- Rod ejection
- Steam generator tube failure
- Loss-of-coolant accidents

With the exception of one analysis, the Catawba 2 Cycle 7 thermal-hydraulic and physics parameters are bounded by the existing CNS FSAR Chapter 15 analyses. In addition, the post-LOCA boron precipitation and post-LOCA containment sump pH analyses given in CNS FSAR Chapter 6 have been reanalyzed. The analyses are as follows.

The dropped rod event is reanalyzed with a cycle specific axial flux shape. The axial flux shape calculated for Catawba 2 Cycle 7 resulted in an axial flux shape which is more peaked than that assumed in the current analysis. The results of the reanalysis demonstrate that the existing limiting case is unchanged by the change in axial flux shape, and remains limiting. The reanalysis requires no Technical Specification changes.

The axial blanketed fuel used in this reload requires the allocation of 3.0% DNBR margin for DNB analyses. This DNBR penalty is to account for the potential non-conservative behavior of the axial power distribution generator in VIPRE-01 when compared to SIMULATE power distributions in blanketed fuel assemblies. This penalty applies only to the axial blanketed fuel and leaves 4.4% DNBR margin for SCD transient analyses and 3.7% DNBR margin for non-SCD transient analyses. Table 7-1 provides the DNBR penalties which are assessed against the available margin.

Post-LOCA subcriticality is reanalyzed for Catawba Unit 2 with higher boron concentrations in the refueling water storage tank (RWST) and the cold leg accumulators (CLA), because the post-LOCA subcriticality for Catawba 2 Cycle 7 fails the acceptance criteria with the existing RWST and CLA boron concentrations. Post-LOCA subcriticality is reanalyzed for Unit 2 with an RWST minimum boron concentration of 2175 ppm and a CLA minimum boron concentration of 2000 ppm. The results of the reanalysis demonstrate that the Catawba 2 Cycle 7 core remains subcritical. Based on the reanalysis, the RWST minimum boron concentration limit is increased from 2000 ppm to 2175 ppm, and the CLA minimum boron concentration limit is increased from 1900 ppm to 2000 ppm. In addition, the RWST and CLA maximum boron concentration limits are increased from 2100 ppm to 2275 ppm in order to preserve operating margin. A Technical Specification change which moves these values to the COLR was submitted January 13, 1993. An SER for this submittal is expected to be issued prior to plant startup for Cycle 7. Therefore, Technical Specification changes are not required, these changes will be made to the COLR for Catawba Unit 2 Cycle 7. Changes to the minimum boron concentrations for the RCS, refueling canal and spent fuel storage pool are being made to be consistent with the boron concentration changes in the RWST.

The increase in the RWST and CLA maximum boron concentration limits necessitates a reanalysis of the post-LOCA boron precipitation evaluation. The results of the reanalysis demonstrate that, with the increased RWST and CLA boron concentrations, post-LOCA boron precipitation is prevented with a reduction in the hot leg recirculation initiation time from 9 hours to 7 hours.

The increase in the RWST and CLA maximum boron concentration limits also necessitates a reanalysis of the post-LOCA containment sump pH. The results of the analysis remain within the existing allowable pH range in the Technical Specification Bases. Therefore, the reanalysis requires no Technical Specification changes.

In addition, the positive breakpoint and slope of the f(ΔI) function of the overtemperature delta T (OTΔT) reactor trip function has been reevaluated for the Cycle 7 reload design. The results of the evaluation demonstrate that the current slope of the f(ΔI) function is overly conservative with respect to optimal core operation. This results in an unacceptable decrease in OTΔT margin to trip during plant startup. It is necessary to decrease the current positive f(ΔI) slope from its current value of 2.316% to 1.525%. All existing licensing basis safety analyses for Catawba Unit 2 Cycle 7 remain valid with the new positive f(ΔI) slope of 1.525%. The new slope will be included in the Catawba Unit 2 Cycle 7 COLR following issuance of the SER for the January 13, 1993 amendment submittal.

Technical Specification changes required for Catawba 2 Cycle 7 operation are provided in Section 8 of this report.

Table 7-1

Transient Analysis DNBR Penalties for Mark-BW Fuel

	SCD Analyses	Non-SCD Analyses
Margin included in CHF limit	10.7%	10.0%
DNBR Penalties		
Instrumentation Biases	2.8%	2.8%
Flow Anomaly	0.5%	0.5%
Axial Blankets	3.0%	3.0%
Total DNBR Penalty	6.3%	6.3%
Available Margin Remaining	4.4%	3.7%

8. PROPOSED MODIFICATIONS TO LICENSING BASIS DOCUMENTS

Revisions to the Technical Specifications have been proposed for Cycle 7 operation due to the impact of the Cycle 7 core design on the post-LOCA subcriticality analysis. Table 8-1 presents the Technical Specification changes required for Cycle 7 operation. Revisions to the Core Operating Limits Report (COLR) are limited to numerical values, and do not involve any changes to the list of parameters reported. Note, a Technical Specification change which moves cycle specific values to the COLR was submitted January 13, 1993. An SER for this submittal is expected to be issued prior to plant startup for Cycle 7. Therefore, when this approval is received the changes to the list of parameters in the COLR will be made. Catawba Unit 2 Cycle 7 will be the first Duke Westinghouse unit to include these values in the COLR.