BEFORE THE UNITED STATES

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Omaha Public Power District	)	Docket No. 50-285
(Fort Calhoun Station	)	
Unit No. 1)	)	

APPLICATION FOR AMENDMENT
OF
OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the U. S. Nuclear Regulatory Commission ("the Commission"), Omaha Public Power District, holder of Facility Operating License No. DPR-40, herewith requests that Technical Specification 2.10 of the Technical Specifications set forth in Appendix A to that License be amended to reflect changes necessary for Cycle 14 operation.

The proposed changes in Technical Specifications and a Discussion, Justification and No Significant Hazards Consideration Analysis, which demonstrates that the proposed changes do not involve significant hazards considerations, was appended as Attachments A and B respectively of an Application of Amendment dated November 27, 1991. The attached information provides corrected information for the Discussion of changes included in the Application dated November 27, 1991. The proposed changes in specifications would not authorize any change in the types or any increase in the amounts of effluents or any change in the authorized power level of the facility.

WHEREFORE, Applicant respectfully requests that Section 2.10 of Appendix A to Facility Operating License No. DPR-40 be amended in the form attached as Attachment A to the Application of Amendment dated November 27, 1991. (LIC-91-0320A)

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Omaha Public Power District	)	Docket No. 50-285
(Fort Calhoun Station	)	
Unit No. 1)	)	

AFFIDAVIT

W. G. Gates, being duly sworn, hereby deposes and says that he is the Division Manager - Nuclear Operations of the Omaha Public Power District; that as such he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached information which provides corrected information for the Discussion of changes included in the Application of Amendment dated November 27, 1991 (LIC-91-0320A) concerning changes necessary for Cycle 14 operation; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information, and belief.

W. G. Gates
Division Manager
Nuclear Operations

STATE OF NEBRASKA)
) ss
COUNTY OF DOUGLAS)

Subscribed and sworn to before me, a Notary Public in and for the State of Nebraska on this 6th day of March, 1992.

GENERAL NOTARY-State of Nebraska
JUDITH E. MAGILL
My Comm. Exp. Febr. 27, 1994

Judith E. Magill
Notary Public

A copy of this Application, including its attachments, has been submitted to the Director - Nebraska State Division of Radiological Health, as required by 10 CFR 50.91.

OMAHA PUBLIC POWER DISTRICT

By M. D. Tates
Division Manager
Nuclear Operations

Subscribed and sworn to before me this 6th day of March, 1992.

GENERAL NOTARY-State of Nebraska
JUDITH E. MAGILL
My Comm. Exp. Febr. 27, 1994

Judith E. Magill
Notary Public

TABLE 1

COMPARISON OF ORIGINAL AND REVISED INPUTS

FOR THE CYCLE 14 TRANSIENT, T/H AND SETPOINT ANALYSES

ITEM	CYCLE 14	CYCLE 14 REVISED	CHANGE	CONSERVATIVE
1. Minimum Available Scram Worth With Most Reactive Rod Stuck 100% Out, %$\Delta\rho$				
a) LOCA, LOFA, Loss of AC, Loss of Load, Seized Rotor, and CEA Withdrawal				
HFP / BOC (PDIL)	6.4074	6.4711	Increase	YES
HFP / EOC (PDIL)	7.6272	7.6793	Increase	YES
HFP / BOC (ARO)	6.7244	6.7920	Increase	YES
HFP / EOC (ARO)	7.9573	8.0129	Increase	YES
b) Loss of Feedwater Flow, Excess Load and Steam Generator Tube Rupture				
HFP / BOC (PDIL)	5.7922	5.8545	Increase	YES
HFP / EOC (PDIL)	6.3493	6.3656	Increase	YES
c) Steam Line Break Event				
HFP / BOC (PDIL)	6.2885	6.3523	Increase	YES
HFP / EOC (PDIL)	7.0243	7.0765	Increase	YES
HFP / BOC (ARO)	6.6055	6.6732	Increase	YES
HFP / EOC (ARO)	7.3545	7.4101	Increase	YES
HZP / BOC (PDIL)	5.0596	5.1035	Increase	YES
HZP / EOC (PDIL)	5.9833	5.9882	Increase	YES
HZP / BOC (ARO)	6.2280	6.2942	Increase	YES
HZP / EOC (ARO)	7.2823	7.3020	Increase	YES

TABLE 1

(continued)

COMPARISON OF ORIGINAL AND REVISED INPUTS

FOR THE CYCLE 14 TRANSIENT, T/H AND SETPOINT ANALYSES

ITEM	CYCLE 14	CYCLE 14 REVISED	CHANGE	CONSERVATIVE
1. Minimum Available Scram Worth With Most Reactive Rod Stuck 100% Out, %$\Delta\rho$ (Continued)				
d) All HZP Events Except CEA Ejection				
HZP / BOC (PDIL)	5.0596	5.1035	Increase	YES
HZP / EOC (PDIL)	5.9833	5.9882	Increase	YES
HZP / BOC (ARO)	6.2280	6.2942	Increase	YES
HZP / EOC (ARO)	7.2823	7.3020	Increase	YES
e) CEA Ejection				
HFP / BOC (PDIL)	5.7879	NOT NECESSARY TO RECALCULATE THESE VALUES DUE TO CEA EJECTION ANALYSIS CONSERVATISMS		
HFP / EOC (PDIL)	6.5936			
HZP / BOC (PDIL)	4.3085			
HZP / EOC (PDIL)	4.8242			

TABLE 1

(continued)

COMPARISON OF ORIGINAL AND REVISED INPUTS FOR THE CYCLE 14 TRANSIENT, T/H AND SETPOINT ANALYSES

ITEM	CYCLE 14 License Application	CYCLE 14 REVISED	CHANGE	CONSERVATIVE
2. Radial Peaking Factors				
a) Planar Radial Peaking Factor (F_{xy})				
ARO	1.8494	1.8386	Decrease	YES
Bank 4	1.9916	1.9924	Increase	NO
Banks 4+3	2.0155	2.0126	Decrease	YES
b) Integrated Radial Peaking Factor (F_r)				
ARO	1.7831	1.7856	Increase	NO
Bank 4	1.9053	1.9122	Increase	NO
Banks 4+3	1.9229	1.9233	Increase	NO

TABLE 1

(continued)

COMPARISON OF ORIGINAL AND REVISED INPUTS

FOR THE CYCLE 14 TRANSIENT, T/H AND SETPOINT ANALYSES

ITEM		CYCLE 14	CYCLE 14 REVISED	CHANGE	CONSERVATIVE
3. CEA Withdrawal Data					
Maximum Differential Worth, $\Delta\rho$/in.		3.5133E−04	3.6059E−04	Increase	**NO**
4. CEA Drop Data					
a) 3−D / HFP, BOC (Biases and Uncertainties Included)					
ARO	Distortion Factor	1.1915	1.1822	Decrease	YES
	Rod Worth, %$\Delta\rho$	0.2947	0.2871	Decrease	**NO**
Bank 4 (PDIL)	Distortion Factor	1.1884	1.1796	Decrease	YES
	Rod Worth, %$\Delta\rho$	0.2940	0.2863	Decrease	**NO**
b) 3−D / 100% Power, 500 MWD/MTU (Biases and Uncertainties Included)					
ARO	Distortion Factor	1.1937	1.1818	Decrease	YES
	Rod Worth, %$\Delta\rho$	N/A	0.2887		
Bank 4 (PDIL)	Distortion Factor	1.1904	1.1812	Decrease	YES
	Rod Worth, %$\Delta\rho$	N/A	0.2880		
c) 3−D / 62.6% Power, BOC (Biases and Uncertainties Included)					
ARO	Distortion Factor	1.2330	1.2240	Decrease	YES
	Rod Worth, %$\Delta\rho$	0.2628	0.2548	Decrease	**NO**
Bank 4 (PDIL)	Distortion Factor	1.2243	N/A		
	Rod Worth, %$\Delta\rho$	0.2634	N/A		
d) 3−D / HFP, EOC (Biases and Uncertainties Included)					
ARO	Distortion Factor	1.1570	1.1495	Decrease	YES
	Rod Worth, %$\Delta\rho$	0.3461	0.3361	Decrease	**NO**
Bank 4 (PDIL)	Distortion Factor	1.1549	N/A		
	Rod Worth, %$\Delta\rho$	0.3424	N/A		

ATTACHMENT 1

2.0 OPERATING HISTORY OF CYCLE 13

Fort Calhoun Station is presently operating in its thirteenth fuel cycle utilizing Batches L, M, N and P fuel assemblies. Fort Calhoun Cycle 13 operation began when criticality was achieved on May 25, 1990, and full power reached on June 18, 1990. The reactor has operated up to the present time with the core reactivity, power distributions, and peaking factors having closely followed the calculated predictions.

Actual Cycle 13 termination burnup of 15,248.82 MWD/MTU was achieved on February 1 1992, and was used as a basis for the reanalysis of Cycle 14.

3.0 GENERAL DESCRIPTION (Continued)

The fuel rod and poison rod locations in Batches M and N shimmed assemblies are shown in Figure 3-2. Figure 3-3 shows the fuel rod locations in Batches N and P unshimmed assemblies. The fuel and poison rod locations for Batch P shimmed assemblies with the fuel rod zone loading technique are shown in Figure 3-4. Due to the Fort Calhoun fuel assembly design, the fuel rods surrounding the five large water holes produce the highest power peaking factors within an assembly. The fuel rod zone loading technique lowers the initial enrichment of U-235 in those fuel rods while maintaining an assembly average initial enrichment sufficient to achieve the Cycle 14 design exposure. Figure 3-5 shows the fuel rod locations for the Batch R1 natural uranium assemblies. Figures 3-6 through 3-9 provide a diagram of each type of fresh assembly which contains IFBA rods.

The average initial enrichment of the 52 fresh Batch R assemblies is 3.57 w/o U-235, a reduction of 0.09 w/o from Cycle 13. Excluding the four fresh natural uranium assemblies, th[illegible]ge initial enrichment is 3.81 w/o U-235. For the second consecutive cyc[illegible]el assembly zone loading technique is used to lower the radial power peaking fa[illegible]in Batches R2 through R7. Batch R2 through R5 assemblies have fuel rods at both 4.[illegible] w/o enriched U-235 and 3.5 w/o enriched U-235, while Batch R6 and R7 assemblies have fuel rods at both 3.75 w/o enriched U-235 and 3.25 w/o enriched U-235.

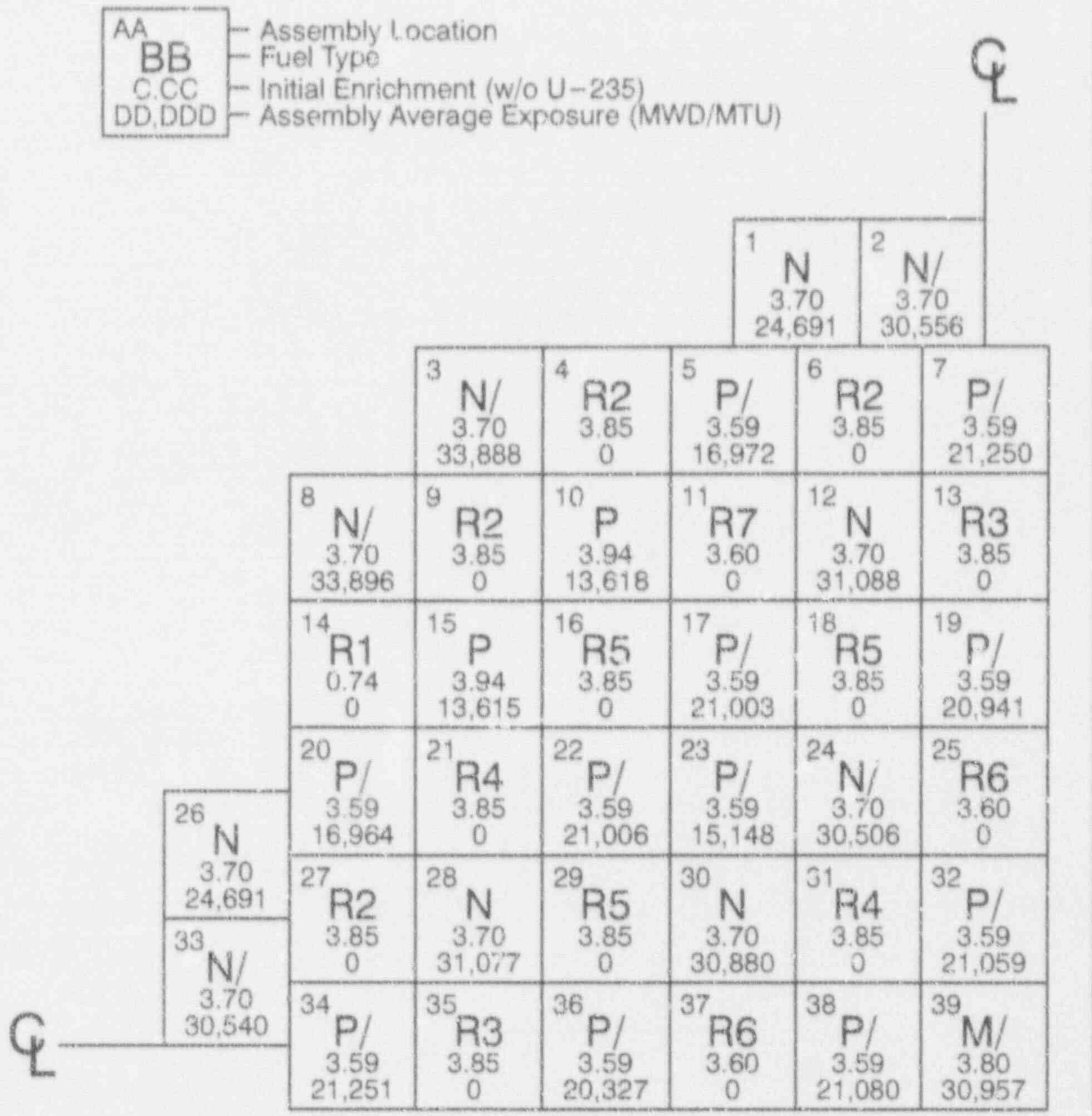
Figure 3-10 shows the beginning of Cycle 14 assembly burnup distribution for a Cycle 13 termination burnup of 15,250 MWD/MTU. The initial enrichment of each fuel assembly is also shown in Figure 3-10. Figure 3-11 shows the projected end of Cycle 14 assembly burnup distribution. The end of Cycle 14 core average exposure will be approximately 28,547 MWD/MTU.

TABLE 3-1

FORT CALHOUN UNIT NO. 1
CYCLE 14 CORE LOADING

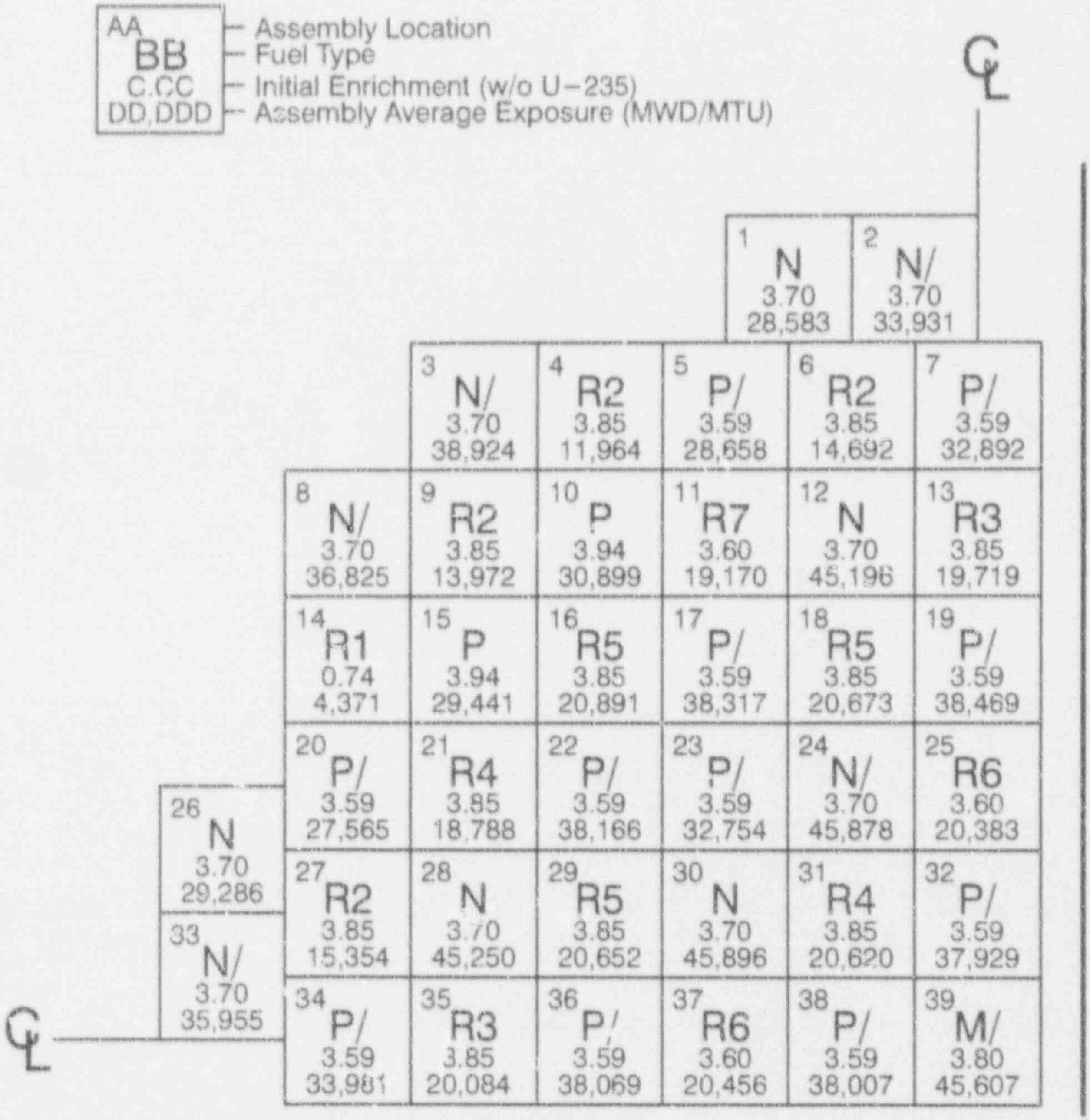
Assembly Designation	Number of Assemblies	BOC Avg. Burnup $^{MWD}/_{MTU}$	EOC Avg. Burnup* $^{MWD}/_{MTU}$	Poison Rods per Assembly	IFBA Rods per Assembly	Initial Poison Loading gm B_{10}/in.
M/	1	31,408	46,196	8	–	0.024
N	20	28,803	38,932	0	–	0
N/	20	32,229	38,574	8	–	0.020
P	8	13,702	30,342	0	–	0
P/	32	19,298	34,464	8	–	0.027
R1	4	0	4,421	–	0	0.003
R2	16	0	13,868	–	28	0.003
R3	4	0	19,962	–	48	0.003
R4	8	0	19,885	–	64	0.003
R5	12	0	20,891	–	84	0.003
R6	4	0	20,598	–	84	0.003
R7	4	0	19,112	–	64	0.003

* Assumes EOC14=14,000 $^{MWD}/_{MTU}$



Note: EOC 13 Burnup = 15,250 MWD/MTU

Cycle 14 BOC Initial Enrichment and Assembly Average Exposure	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3-10



Cycle 14 EOC Initial Enrichment and Assembly Average Exposure	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3-11

5.0 NUCLEAR DESIGN

5.1 PHYSICAL CHARACTERISTICS

5.1.1 Fuel Management

The Cycle 14 fuel management uses an extreme low radial leakage design, with twice burned assemblies predominantly loaded on the periphery of the core with hafnium flux suppression rods inserted into the guide tubes of selected peripheral fuel assemblies adjacent to the reactor vessel limiting welds. This extreme low radial leakage fuel loading pattern is utilized to minimize the flux to the pressure vessel welds and achieve the maximum in neutron economy. Use of this type of fuel management to achieve reduced pressure vessel flux over a standard out–in–in pattern results in higher radial peaking factors. The maximum radial peaking factors for Cycle 14 have been reduced by lowering the enrichment of the fuel pins adjacent to the fuel assembly water holes as described in Section 3.0.

Also described in Section 3.0 is the Cycle 14 loading pattern which is composed of 52 fresh Batch R assemblies of which 48 contain the aforementioned IFBA pellet design. The remaining 4 Batch R assemblies contain fuel rods that are loaded with naturally enriched uranium and also placed in locations near the limiting welds. All of these 48 assemblies employ intra–assembly uranium enrichment splits. Batches R2 through R5 contain a high pin enrichment of 4.00 w/o and a low pin enrichment of 3.50 w/o, Batches R6 and R7 contain a high pin enrichment of 3.75 w/o and a low pin enrichment of 3.25 w/o. Forty twice burned N assemblies are being returned to the core, along with 40 once burned P assemblies. One twice burned M assembly, which was discharged into the spent fuel pool at the end of Cycle 12, will be returned to the core and used as the center assembly. This assembly arrangement will produce a Cycle 14 loading pattern with a cycle energy of 14,000 MWD/MTU with an additional 1,000 MWD/MTU of energy in a coastdown mode if required. The Cycle 14 core characteristics have been examined for a Cycle 13 termination of 15,250 MWD/MTU and limiting values established for the safety analysis.

Physics characteristics including reactivity coefficients for Cycle 14 are listed in Table 5–1 along with the corresponding values from Cycle 13. It should be noted that the values of parameters actually employed in the safety analyses are different from those displayed in Table 5–1 and are typically chosen to conservatively bound predicted values with accommodation for appropriate uncertainties and allowances.

The BOC, HZP conditions for all events are the most limiting conditions used in the determination of available shutdown margin for compliance with the Technical Specifications. The minimum available shutdown margin is 1.06%$\Delta\rho$ with respect to the Technical Specification limit of

5.0 NUCLEAR DESIGN (Continued)

5.1 PHYSICAL CHARACTERISTICS (Continued)

5.1.1 Fuel Management (Continued)

4.0%$\Delta\rho$. Table 5-2 presents a summary of CEA shutdown worths and reactivity allowances for Cycle 14. The cycle 14 CEA worth values, used in the calculation of minimum scram worth, exceed the minimum value required by Technical Specifications and thus provide an adequate shutdown margin.

5.1.2 Power Distribution

Figures 5-1 through 5-3 illustrate the all rods out (ARO) planar radial power distributions at BOC14, MOC14, and EOC14, respectively, and are based upon the Cycle 13 late window burnup timepoint. These radial power densities are assembly averages representative of the entire core length. The high burnup end of the Cycle 13 shutdown window tends to increase the power peaking in the high power assemblies in the Cycle 14 fuel loading pattern. The radial power distributions, with Bank 4 fully inserted at beginning and end of Cycle 14, are shown in Figures 5-4 and 5-5, respectively.

The radial power distributions described in this section are calculated data without uncertainties or other allowances with the exception of the single rod power peaking values. For both DNB and kW/ft safety and setpoint analyses in either rodded or unrodded configurations, the power peaking values actually used are higher than those expected to occur at any time during Cycle 14. These conservative values, which are used in Section 7.0 of this document, establish the allowable limits for power peaking to be observed during operation.

As previously indicated, Figures 3-5 and 3-6 show the integrated assembly burnup values at 0 and 14,000 MWD/MTU for Cycle 14.

The range of allowable axial peaking is defined by the limiting conditions for operation and their axial shape index (ASI). Within these ASI limits, the necessary DNBR and kW/ft margins are maintained for a wide range of possible axial shapes. The maximum three-dimensional or total peaking factor (F_q) anticipated in Cycle 14 during normal base load, all rods out operation at full power is 2.1069, including uncertainty allowances.

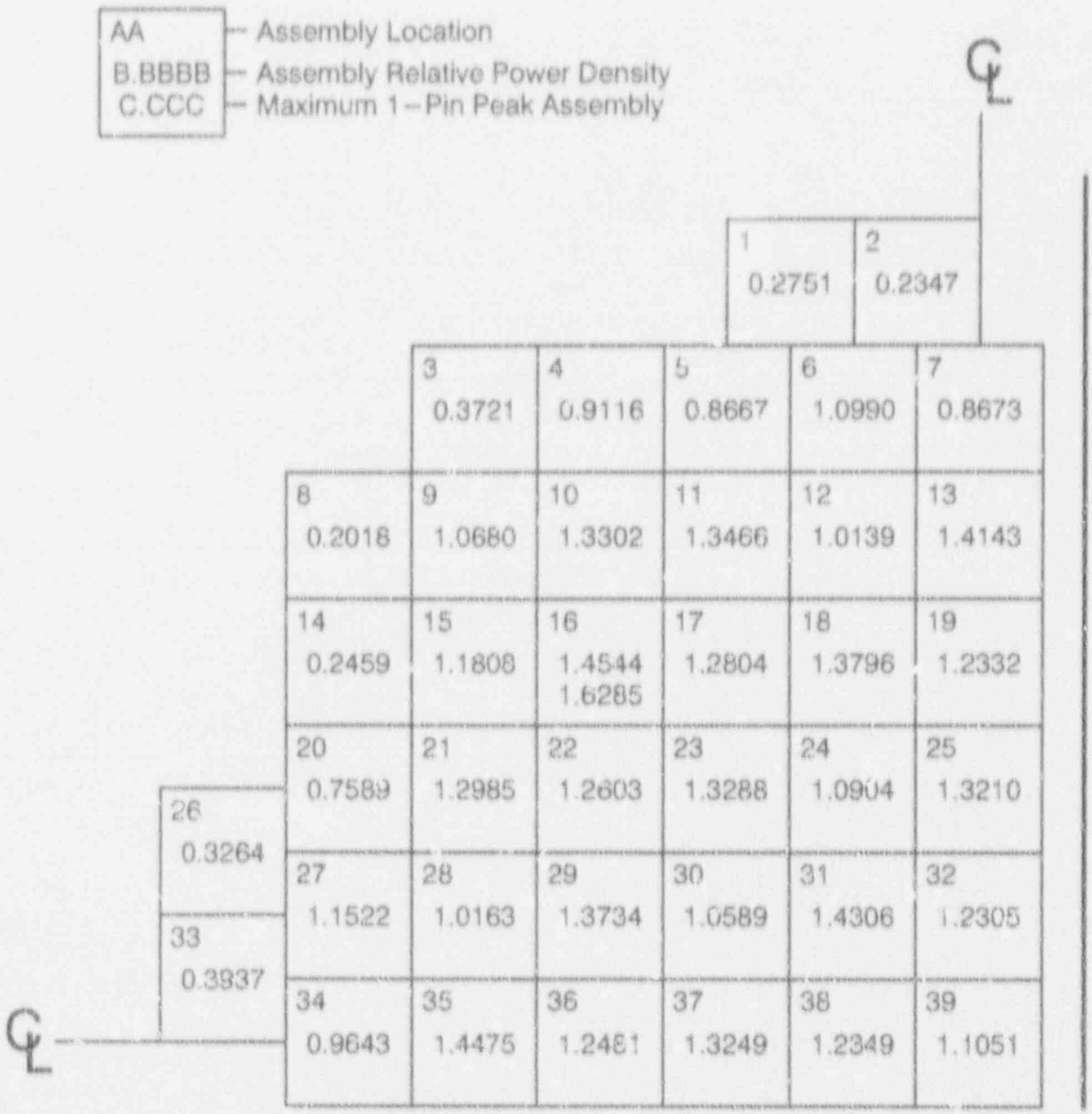
TABLE 5-2

FORT CALHOUN UNIT NO. 1, CYCLE 14
LIMITING VALUES OF REACTIVITY WORTHS AND ALLOWANCES**
FOR HOT ZERO POWER

		BOC, HZP ($\%\Delta\rho$)	EOC, HZP ($\%\Delta\rho$)
1.	Worth of all CEAs Inserted	7.52	8.86
2.	Stuck CEA Allowance	1.17	1.43
3.	Worth of all CEAs Less Worth of Most Reactive CEA Stuck Out	6.35	7.43
4.	Power Dependent Insertion Limit CEA Worth	1.19	1.33
5.	Calculated Scram Worth	5.16	6.10
6.	Physics Uncertainty plus Bias	0.10*	0.12*
7.	Net Available Scram Worth	5.06	5.98
8.	Technical Specification Shutdown Margin	4.00	4.00
9.	Margin in Excess of Technical Specification Shutdown Margin	1.06	1.98

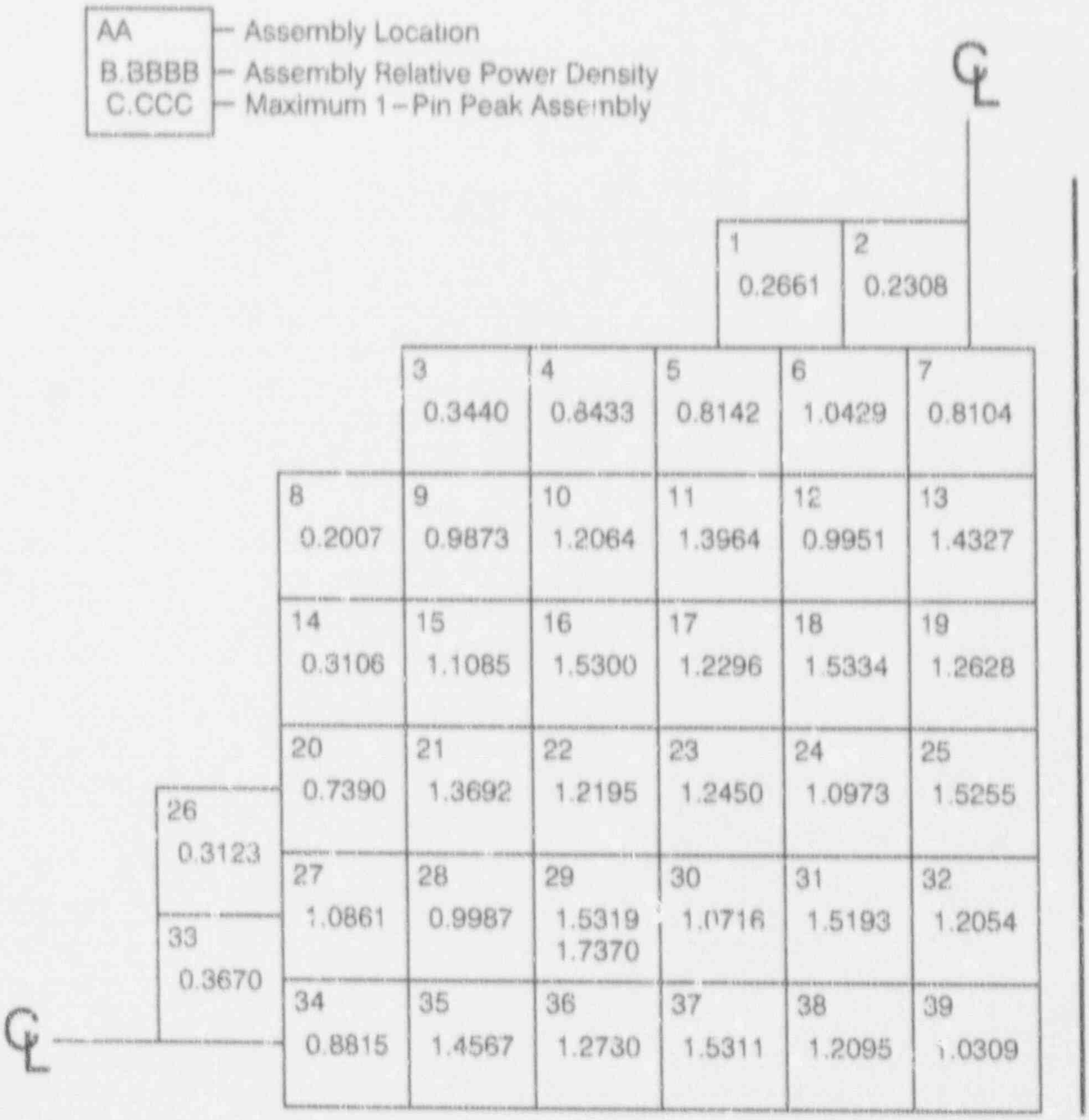
* 1.96% of calculated scram worth from revised ABB-CE methodology biases and uncertainties.

** These values are the same values as the original analysis, prior to detection and correction of the hafnium-related cross-section error. The results remain conservative with respect to the corrected scram worths, i.e. the above values are less than the revised values.



Maximum 1-Pin Peak at 23% Core Height

Cycle 14 Assembly Power Distribution 0 MWD/MTU, HFP, Equilibrium Xenon	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 5-1



Maximum 1-Pin Peak at 23% Core Height

Cycle 14 Assembly Power Distribution 7,000 MWD/MTU, HFP, Eq. Xenon	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 5-2

AA	Assembly Location
B.BBBB	Assembly Relative Power Density
C.CCC	Maximum 1-Pin Peak Assembly

Assembly Location	Assembly Relative Power Density	Maximum 1-Pin Peak Assembly
1	0.3175	
2	0.2775	
3	0.3963	
4	0.9056	
5	0.8645	
6	1.1034	
7	0.8602	
8	0.2426	
9	1.0402	
10	1.1908	
11	1.3758	
12	0.9966	
13	1.3986	
14	0.4078	
15	1.1243	
16	1.4778	1.6550
17	1.1698	
18	1.4593	
19	1.1899	
20	0.8223	
21	1.3852	
22	1.1702	
23	1.1710	
24	1.0403	
25	1.4233	
26	0.3851	
27	1.1709	
28	1.0130	
29	1.4674	
30	1.0194	
31	1.4047	
32	1.1143	
33	0.4494	
34	0.9477	
35	1.4310	
36	1.2082	
37	1.4289	
38	1.1175	
39	0.9687	

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Maximum 1-Pin Peak at 17% Core Height

Cycle 14 Assembly Power Distribution 14,000 MWD/MTU, HFP, Eq. Xenon	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 5-3

AA	Assembly Location
B.BBBB	Assembly Relative Power Density
C.CCC	Maximum 1-Pin Peak Assembly

					1 0.2949	2 0.2571
		3 0.2572	4 0.8059	5 0.8772	6 1.1819	7 0.9490
	8 0.1284	9 0.5366	10 1.1595	11 1.3600	12 1.0870	13 1.5387
	14 0.2191	15 1.0341	16 1.3964	17 1.3206	18 1.4719	19 1.3286
26 0.3633	20 0.7986	21 1.3397	22 1.3099	23 1.3957	24 1.1409	25 1.3719
	27 1.2728	28 1.1100	29 1.4798	30 1.1114	31 1.4117	32 1.1376
33 0.4444	34 1.0791	35 1.6002 1.7810	36 1.3582	37 1.3824	38 1.1438	39 0.6552

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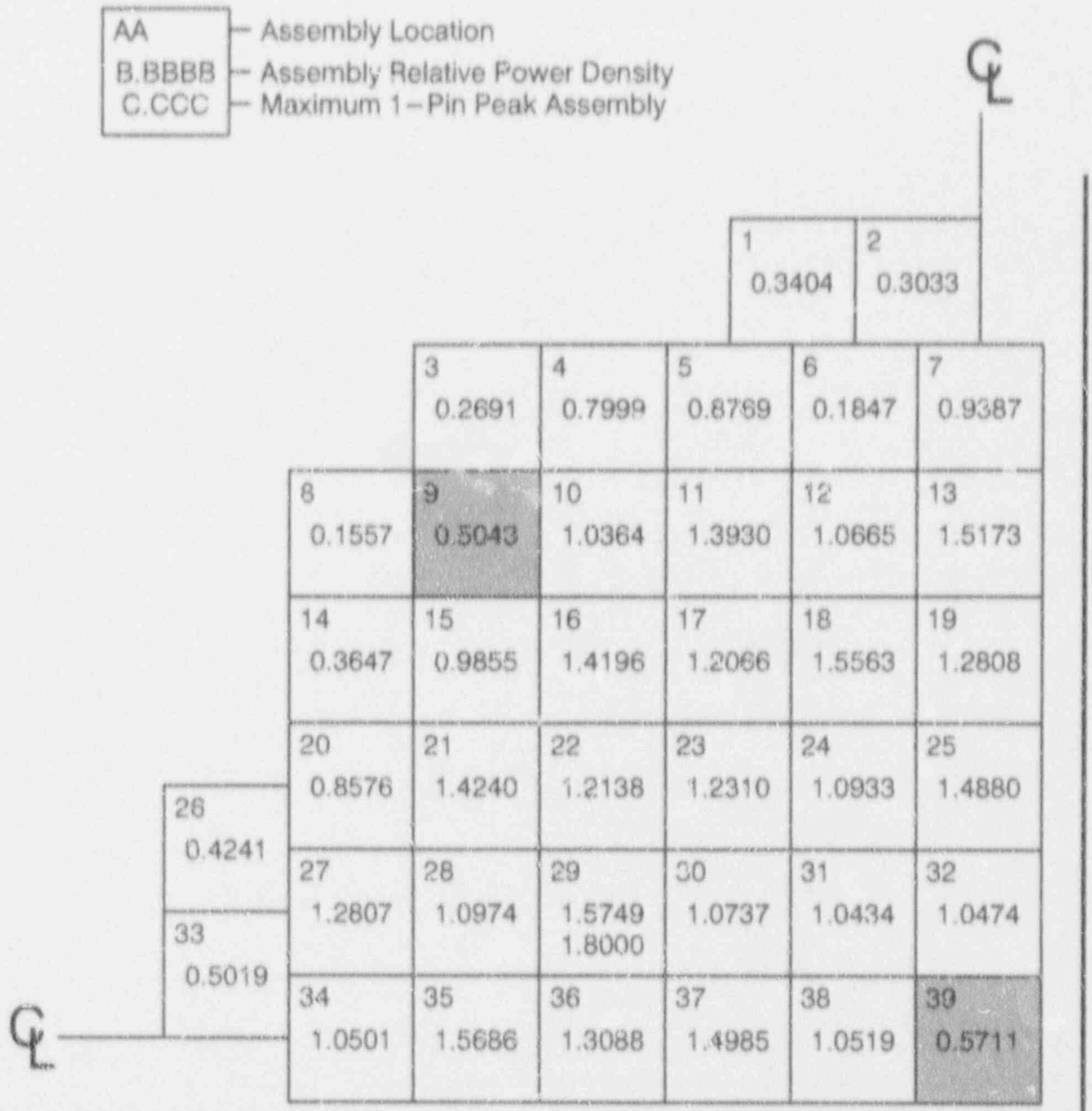
Maximum 1-Pin Peak at 20% Core Height

– Bank 4 Locations

Cycle 14 Assembly RPD Bank 4 In 0 MWD/MTU, HFP, Equilibrium Xenon	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 5-4



Maximum 1-Pin Peak at 17% Core Height

- Bank 4 Locations

Cycle 14 Assembly RPD Bank 4 In 14,000 MWD/MTU, HFP, Eq. Xenon	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 5-5

6.0 THERMAL-HYDRAULIC DESIGN

6.1 DNBR ANALYSIS

Steady state DNBR analyses of Cycle 14 at the rated power of 1500 MWt have been performed using the TORC computer code described in Reference 1 and the CE-1 critical heat flux correlation described in Reference 2. The CETOP-D computer code described in Reference 3 was used in the setpoint analysis, but was replaced by the TORC code for DNBR analyses. The DNBR analysis applications and methods did not change from previous cycles, with the exception that the TORC computer code was used to calculate the minimum DNBR rather than the CETOP-D computer code. Both codes are approved for use with the OPPD methods. This is different from the combination that was used in the Cycle 8 through Cycle 13 Fort Calhoun reload analyses (References 4 through 9). The reload methodology for Cycle 14 can be found in Reference 10.

Table 6-1 contains a list of pertinent thermal-hydraulic parameters used in both safety analyses and for generating reactor protective system setpoint information. The calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) listed in Table 6-1 have been combined statistically with other uncertainty factors at the 95/95 confidence/probability level (Reference 11) to define the design limit on CE-1 minimum DNBR.

6.2 FUEL ROD BOWING

The fuel rod bow penalty accounts for the adverse impact on MDNBR of random variations in spacing between fuel rods. The penalty at 45,000 MWD/MTU burnup is 0.5% in MDNBR. This penalty was applied in the derivation of the SCU MDNBR design limit of 1.18 (References 6 and 12) in the statistical combination of uncertainties (Reference 11). The design basis for the amount of fuel rod bow allowed in the Westinghouse fuel and for the CE fuel design is the same. Westinghouse has identified in the mechanical fuel design report that the amount of deflection does not require a DNB penalty to be applied under Westinghouse analysis requirements. Thus, the CE DNB penalty was applied to the Westinghouse fuel to ensure that the OPPD statistical combination of uncertainties were still valid and that conservative input assumptions were used in the analysis.

TABLE 6-1

FORT CALHOUN UNIT NO. 1, CYCLE 14
THERMAL HYDRAULIC PARAMETERS AT FULL POWER

	Unit	Cycle 14*
Total Heat Output (Core Only)	MWt 10^6 BTU/hr	1500 5119
Fraction of Heat Generated in Fuel Rod		0.975
Primary System Pressure		
Nominal	psia	2100
Minimum In Steady State	psia	2075
Maximum In Steady State	psia	2150
Inlet Temperature (Maximum)	°F	545
Total Reactor Coolant Flow	gpm	202,500
(Steady State)	10^6 lbm/hr	76.32
(Through the Core)	10^6 lbm/hr	73.06
Hydraulic Diameter (Nominal Channel)	ft	.044
Average Mass Velocity	10^6 lbm/hr-ft^2	2.226
Core Average Heat Flux (Accounts for Heat Generated in Fuel Rod)	BTU/hr-ft^2	181281
Total Heat Transfer Surface Area	ft^2	28,241**
Average Core Enthalpy Rise	BTU/lbm	72.6
Average Linear Heat Rate	kW/ft	6.01**
Engineering Heat Flux Factor		1.03***
Engineering Factor on Hot Channel Heat Input		1.03***
Rod Pitch and Bow		1.065***
Fuel Densification Factor (Axial)		1.002

* Design inlet temperature and nominal primary system pressure were used to calculate these parameters.

** Based on Cycle 14 specific value of 424 fuel displacing shims.

*** These factors were combined statistically (Reference 8) with other uncertainty factors at 95/95 confidence/probability level to define a design limit on CE-1 minimum DNBR.

TABLE 7-1

FORT CALHOUN UNIT NO. 1, CYCLE 14
DESIGN BASIS EVENTS CONSIDERED IN THE NON-LOCA SAFETY ANALYSIS

7.1 Anticipated Operational Occurrences for which intervention of the RPS is necessary to prevent exceeding acceptable limits:

7.1.1	Reactor Coolant System Depressurization	Reanalyzed
7.1.2	Loss of Load	Not Reanalyzed[5]
7.1.3	Loss of Feedwater Flow	Not Reanalyzed[5]
7.1.4	Excess Heat Removal due to Feedwater Malfunction	Not Reanalyzed[5]
7.1.5	Startup of an Inactive Reactor Coolant Pump	Not Reanalyzed[1]

7.2 Anticipated Operational Occurrences for which sufficient initial steady state thermal margin, maintained by the LCOs, is necessary to prevent exceeding the acceptable limits:

7.2.1	Excess Load	Reanalyzed[2]
7.2.2	Sequential CEA Group Withdrawal	Reanalyzed[2]
7.2.3	Loss of Coolant Flow	Reanalyzed[3]
7.2.4	CEA Drop	Reanalyzed
7.2.5	Boron Dilution	Reviewed
7.2.6	Transients Resulting from the Malfunction of One Steam Generator	Not Reanalyzed[4]

7.3 Postulated Accidents

7.3.1	CEA Ejection	Reanalyzed
7.3.2	Steam Line Break	Reviewed[5]
7.3.3	Seized Rotor	Reanalyzed[5]
7.3.4	Steam Generator Tube Rupture	Not Reanalyzed

1 Technical Specifications preclude this event during operation.

2 Requires High Power and Variable High Power Trip.

3 Requires Low Flow Trip.

4 Requires trip on high differential steam generator pressure.

5 Event bounded by reference cycle analysis. A negative determination utilizing the 10 CFR 50.59 criteria was made for this event.

TABLE 7-2

FORT CALHOUN UNIT NO. 1, CYCLE 14
CORE PARAMETERS INPUT TO SAFETY ANALYSES
FOR DNB AND CTM (CENTERLINE TO MELT) DESIGN LIMITS

Physics Parameters	Units	Cycle 13 Values	Cycle 14 Values
Radial Peaking Factors			
For DNB Margin Analyses (F_R^T)			
Unrodded Region		1.70*	1.79*
Bank 4 Inserted		1.73*	1.92*
For Planar Radial Component (F_{xy}^T) of 3-D Peak (CTM Limit Analyses)			
Unrodded Region		1.75*	1.85*
Bank 4 Inserted		1.77*	2.0*
Maximum Augmentation Factor		1.000	1.000
Moderator Temperature Coefficient	$10^{-4}\ \Delta\rho/°F$	-2.7 to +0.5	-3.0 to +0.5
Shutdown Margin (Value Assumed in Limiting EOC Zero Power SLB)	$\%\Delta\rho$	-4.0	-4.0

* The DNBR analyses utilized the methods discussed in Section 6.1 of this report. The procedures used in the Statistical Combination of Uncertainties (SCU) as they pertain to DNB and CTM limits are detailed in References 2-5.

TABLE 7-3

FORT CALHOUN UNIT NO. 1
DESIGN BASIS EVENTS REANALYZED FOR CYCLE 14

Event	Reason for Reanalysis	Acceptance Criteria	Summary of Results
Sequential CEA Group Withdrawal	Calculate cycle specific ROPM values	Minimum DNBR ≥ 1.18 using the CE-1 correlation. Transient PLHGR ≤ 22 kW/ft.	MDNBR =1.71 PLHGR < 22 kW/ft
CEA Drop	Incorporated bounding input values	Minimum DNBR ≥ 1.18 using CE-1 correlation. Transient PLHGR ≤ 22 kW/ft	MDNBR = 1.38 PLHGR < 22 kW/ft
Excess Load	Reclassified as a ROPM event (methodology change)	Minimum DNBR ≥ 1.18 using CE-1 correlation. Transient PLHGR ≤ 22 kW/ft	MDNBR = 1.31 PLHGR < 22 kW/ft
RCS Depressurization	To provide a conservative Pbias input for the TM/LP due to the Excess Load methodology change	Pbias value ≤ the previous cycle's limiting value (from Excess Load and RCS Depressurization)	Pbias = 30 psia
Loss of Coolant Flow	To provide for incruased instrument uncertainty in the RPS low flow trip circuit. This would reduce the trip setpoint to 90% of full flow conditions.	Minimum DNBR ≥ 1.18 using CE-1 correlation. Transient PLHGR ≤ 22 kW/ft	MDNBR = 1.42 PLHGR < 22 kW/ft

7.0 TRANSIENT ANALYSIS (Continued)

7.2 ANTICIPATED OPERATIONAL OCCURRENCES (CATEGORY 2)

7.2.2 CEA Withdrawal Event

The CEA Withdrawal (CEAW) event was reanalyzed for Cycle 14 to determine the initial margins that must be maintained by the Limiting Conditions for Operations (LCOs) such that the DNBR and fuel centerline to melt (CTM) design limits will not be exceeded in conjunction with the RPS (Variable High Power, High Pressurizer Pressure, or Axial Power Distribution Trips).

The methodology contained in Reference 6 was employed in analyzing the CEAW event. This event is classified as one for which the acceptable DNBR and CTM limits are not violated by virtue of maintenance of sufficient initial steady state thermal margin provided by the DNBR and Linear Heat Rate (LHR) related LCOs.

For the HFP CEAW DNBR analysis, a MTC value identical to that utilized in Reference 8 and a gap thermal conductivity consistent with the assumption of Reference 6 were used in conjunction with a variable reactivity insertion rate.

The HFP case for Cycle 14 is considered to meet the 10 CFR 50.59 criteria since the results show that the required overpower margin is less than the available overpower margin required by the Technical Specifications for the DNB and PLHGR LCOs. Since a negative 10 CFR 50.59 determination was made for Cycle 14, the conclusions for Cycle 12 remain valid and applicable to Cycle 14.

The zero power case was analyzed to demonstrate that acceptable DNBR and centerline melt limits are not exceeded. For the zero power case, a reactor trip, initiated by the Variable High Power Trip at 30% (20% plus 10% uncertainty of rated thermal power) was assumed in the analysis.

The 10 CFR 50.59 criteria are satisfied for the HZP event if the minimum DNBR is greater than that reported in the reference cycle.

The zero power case initiated at the limiting conditions of operation results in a a minimum CE−1 DNBR of 5.44 which is less than the Cycle 12 value of 6.99, but still far in excess of the minimum 1.18 DNBR limit. The analysis shows that the fuel to centerline melt temperatures are well below those corresponding to the acceptable fuel to centerline melt limit. The key input parameters used for the zero power case are presented in Table 7.2.2−1.

TABLE 7.2.2-1

FORT CALHOUN UNIT NO. 1, CYCLE 14
KEY PARAMETERS ASSUMED IN THE HZP CEA WITHDRAWAL ANALYSIS

Parameter	Units	Cycle 12	Cycle 14
Initial Core Power Level	MWt	1	1*
Core Inlet Coolant Temperature	°F	532	532*
Pressurizer Pressure	psia	2053	2075*
Moderator Temperature Coefficient	$\times 10^{-4}\ \Delta\rho/°F$	+0.5	+0.5
Doppler Coefficient Multiplier		0.85	0.85
CEA Worth at Trip	$\%\Delta\rho$	5.28	5.048
Reactivity Insertion Rate Range	$\times 10^{-4}\ \Delta\rho/sec$	0 to 1.0	0 to 2.7
CEA Group Withdrawal Rate	in/min	46	46
Holding Coil Delay Time	sec	0.5	0.5

* The DNBR calculations used the methods discussed in Section 6.1 of this document and detailed in References 2 through 5. The effects of uncertainties on these parameters were accounted for statistically in the DNBR and CTM calculations.

7.0 TRANSIENT ANALYSIS (Continued)

7.2 ANTICIPATED OPERATIONAL OCCURRENCES (CATEGORY 2) (Continued)

7.2.3 Loss of Coolant Flow Event

The Loss of Coolant Flow event was reanalyzed for Cycle 14 to determine the minimum initial margin that must be maintained by the Limiting Conditions for Operations (LCOs) such that in conjunction with the RPS low flow trip, the DNBR limit will not be exceeded.

The event was analyzed parametrically in initial axial shape and rod configuration using the methods described in Reference 6 (which utilizes the statistical combination of uncertainties in the DNBR analysis as described in Appendix C of References 4 and 5).

The 4-Pump Loss of Coolant Flow produces a rapid approach to the DNBR limit due to the rapid decrease in the core coolant flow. Protection against exceeding the DNBR limit for this transient is provided by the initial steady state thermal margin which is maintained by adhering tho the LCOs on DNBR margin and by the response of the RPS which provides an automatic reactor trip on low reactor coolant flow as measured by the steam generator differential pressure transmitters.

The flow coastdown is generated by the CESEC-III (References 9 and 10) which utilizes implicit modeling of the reactor coolant pumps. Table 7.2.3-1 lists the key transient parameters used in the Cycle 14 analysis and compares them to the reference cycle (Cycle 12) values.

The low flow trip setpoint is reached at 3.66 seconds and the scram rods start dropping into the core 1.15 seconds later. A minimum CE-1 DNBR of 1.422 is reached at 5.5 seconds.

It may be concluded that for Cycle 14 the Loss of Flow event, when initiated from the LCOs, in conjunction with the Low Flow Trip, will not exceed the minimum DNBR design limit.

TABLE 7.2.3-1

FORT CALHOUN UNIT NO.1, CYCLE 14
KEY PARAMETERS ASSUMED IN THE LOSS OF COOLANT FLOW ANALYSIS

Parameter	Units	Cycle 12	Cycle 14
Initial Core Power Level	MWt	1500*	1500*
Initial Core Inlet Coolant Temperature	°F	545*	545*
Initial RCS Flow Rate	gpm	208,280*	202,500*
Pressurizer Pressure	psia	2075*	2075*
Moderator Temperature Coefficient	$\times 10^{-4}\ \Delta\rho/°F$	+0.5	+0.5
Doppler Temperature Multiplier		0.85	0.85
CEA Worth at Trip (ARO)	$\%\Delta\rho$	-6.50	-6.72
LFT Analysis Setpoint	% of initial flow	93	90
LFT Response Time	sec	0.65	0.65
CEA Holding Coil Delay	sec	0.5	0.5
CEA Time to 100% Insertion (Including Holding Coil Delay	sec	3.1	3.1
Total Unrodded Radial Peaking Factor (F_R^T)		1.80	1.79

* The uncertainties on these parameters were combined statistically rather than deterministically. The values listed represent the bounds included in the statistical combination.

TABLE 7.2.4-1

FORT CALHOUN UNIT NO. 1, CYCLE 14
KEY PARAMETERS ASSUMED IN THE HFP CEA DROP ANALYSIS

Parameter	Units	Cycle 11	Cycle 14
Initial Core Power Level	MWt	1500*	1500*
Core Inlet Coolant Temperature	°F	543*	545*
Pressurizer Pressure	psia	2075*	2075*
Core Mass Flow Rate	gpm	202,500*	196,000*
Moderator Temperature Coefficient	$\times 10^{-4}\ \Delta\rho/°F$	-2.7	-3.0
Doppler Coefficient Multiplier		1.15	1.40
CEA Insertion at Maximum Allowed Power	%Insertion of Bank 4	25	25
Dropped CEA Worth	Unrodded, $\%\Delta\rho$ PDIL, $\%\Delta\rho$	-0.2337 -0.2295	-0.2887 -0.2880
Maximum Allowed Power Shape Index at Negative Extreme of LCO Band		-0.18	-0.18
Radial Peaking Distortion Factor	Unrodded Region Bank 4 Inserted	1.1566 1.1598	1.1818 1.1812

* The DNBR calculations used the methods discussed in Section 6.1 of this document and detailed in References 2 through 5. The effects of uncertainties on these parameters were accounted for statistically in the DNBR and CTM calculations.

TABLE 7.2.4-2

FORT CALHOUN UNIT NO.1, CYCLE 14
SEQUENCE OF EVENTS FOR FULL LENGTH CEA DROP

Time (sec)	Event	Setpoint or Value
0.0	CEA Begins to Drop into Core	---
1.0	CEA Reaches Fully Inserted Position	100% Insertion
1.14	Core Power Level Reaches a Minimum and Begins to Return to Power Due to Reactivity Feedbacks	64.4% of 1500 MWt
74.9	Core Inlet Temperature Reaches a Minimum Value	538.85°F
199.5	RCS Pressure Reaches a Minimum Value	1998.24 psia
200.0	Core Power Returns to its Maximum Value	94.98% of 1500 MWt
200.0	Minimum DNBR is Reached	1.379 (CE-1 Correlation)

TABLE 7.2.5-1

FORT CALHOUN UNIT NO. 1, CYCLE 14
KEY PARAMETERS ASSUMED IN THE BORON DILUTION ANALYSIS

Parameters	Cycle 13 Values	Cycle 14 Values
Critical Boron Concentration, ppm (ARO, No Xenon)		
Mode		
Hot Standby	1662	1292
Hot Shutdown	1662	1292
Cold Shutdown – Normal RCS Volume	1457	1204
Cold Shutdown – Minimum RCS Volume*	1279	1204
Refueling	1454	1180
Inverse Boron Worth, ppm/%Δρ		
Mode		
Hot Standby	−90	−90
Hot Shutdown	−55	−55
Cold Shutdown – Normal RCS Volume	−55	−55
Cold Shutdown – Minimum RCS Volume	−55	−55
Refueling	−55	−55
Minimum Shutdown Margin Assumed, %Δρ		
Mode		
Hot Standby	−4.0	−4.0
Hot Shutdown	−4.0	−4.0
Cold Shutdown – Normal RCS Volume	−3.0	−3.0
Cold Shutdown – Minimum RCS Volume*	−3.0	−3.0
Refueling (ppm)**	1900	1700***

* Shutdown Groups A and B out, all Regulating Groups inserted except most reactive rod stuck out.

** Includes a 5.0%Δρ shutdown margin.

*** Proposed Cycle 14 COLR value.

7.0 TRANSIENT ANALYSIS (Continued)

7.3 POSTULATED ACCIDENTS (Continued)

7.3.3 Seized Rotor Event

The Seized Rotor event was reanalyzed for Cycle 14 to demonstrate that only a small fraction of fuel pins are predicted to fail during this event. The analysis showed that Cycle 14 is bounded by the reference cycle (Cycle 9) analysis because an F_R^T of 1.85 was assumed in the Cycle 9 analysis and the Cycle 14 Technical Specification of 1.79 remains conservative with respect to the F_R^T value used in the Cycle 9 analysis.

Therefore, the total number of pins predicted to fail will continue to be less than 1% of all of the fuel pins in the core. Based on this result, the resultant site boundary dose would be well within the limits of 10 CFR 100.

ATTACHMENT 2

The purpose of this attachment is to provide OPPD's justification for the use of the new calculational uncertainties for peaking factors derived from the ROCS-NEM methods.

Use of upgraded reactor physics codes necessitates the use of uncertainties and biases consistent with application of the new methods. It will be shown that peaking factors calculated using the new methods, with biases and uncertainties (consistent with the new methods) applied, are comparable to those obtained using the former methods (with the former biases and uncertainties included).

The NRC-approved reference cycle (i.e. Cycle 13) reload submittal utilized the Higher Order Difference (HOD) method which is described in Reference 1. For the Cycle 14 submittal, the Nodal Expansion Method (NEM), which is also described in Reference 1, was implemented to increase the calculational accuracy of the nuclear design codes. Specific changes incorporating the new methods include:

1. Implementation of NEM into the ROCS code;
2. Improvements in accountability of anisotropic scattering and higher order interface current angular distributions in the DIT code;
3. Introduction of assembly discontinuity factors between the ROCS and DIT codes;
4. Update of biases and uncertainties applied to calculated parameters.

The revised biases and uncertainties associated with the application of the NEM are described in Reference 2. Introduction of the improved methods required the re-evaluation of the biases and uncertainties. The ABB-Combustion Engineering data base used to establish the biases and uncertainties was expanded to reflect recent reload cycles with low leakage and high burnup fuel management. The data base was derived from the following sources which includes Fort Calhoun Station:

Plant	Cycle
Palo Verde 1	2
Palo Verde 2	2
Palo Verde 3	2
Palo Verde 1	3
Palo Verde 2	3
Calvert Cliffs 1	10
Calvert Cliffs 2	8
Fort Calhoun	13

Total Cycles: 8

In addition, Calvert Cliffs 2, Cycle 9 data was added later and found to be consistent with the above data base.

For justification of the application of the NEM biases and uncertainties to the unrodded planar and integrated radial peaking factors, the uncertainty plus bias terms (i.e. upper tolerance limits) appropriate for use are found in Reference 2, Table D (Items D-5 and D-6), page D-1. The NEM upper tolerance limits for pin peaking factors F_{xy} and F_r are 5.35% and 4.00%, respectively. Using the former HOD method (Reference 1), upper tolerance limits for pin peaking factors F_{xy} and F_r are 4.99% and 3.02%, respectively. In order for both NEM and HOD methods to produce similar pin peaking factors, the NEM upper tolerance limits must be larger in value than the HOD upper tolerance limits. Therefore, the pin peaking factor upper tolerance limits for NEM are more conservative than the pin peaking factor upper tolerance limits for HOD.

To verify that use of the HOD method and the NEM method produces similar results, a Cycle 13 pin peaking factor model using NEM was generated and compared to the Cycle 13 HOD results. The results of this comparison, along with the upper tolerance limits for both methods, are presented below along with results from Cycle 14 using NEM:

FORT CALHOUN UNIT NO. 1
MAXIMUM PIN PEAKING FACTORS AND ALLOWANCES

		Cycle 13 (HOD)	Cycle 13 (NEM)	Cycle 14 (NEM)
1. Upper Tolerance Limits (%),				
	F_{xy}	4.99[(1)]	5.35[(2)]	5.35[(2)]
	F_r	3.02[(1)]	4.00[(2)]	4.00[(2)]
2. Calculated Maximum Peaking Factors, (from ROCS)				
	F_{xy}	1.585	1.588	1.745
	F_r	1.553	1.558	1.717
3. Final Maximum Peaking Factors, (Calculated + Upper Tolerance)				
	F_{xy}	**1.664**	**1.673**	1.839
	F_r	**1.600**	**1.621**	1.786

(1) HOD Upper Tolerance Limit from CENPD-266-P-A
(2) NEM Upper Tolerance Limit from CE-CES-129, Revision 1-P

The results show that the differences between the HOD and NEM methods for calculating the Cycle 13 F_{xy} and F_r maximum pin peaking factors to be 0.54% and 1.31%, respectively, which are considered to be acceptably small. It can also be calculated from the above results that NEM produces slightly more conservative maximum pin peaking factors than HOD.

In summary, the application of the revised upper tolerance limits to the pin peaking factors (as presented in Reference 2) are considered to be justified for application with the NEM method rather than the HOD method. Both methods are described in Reference 1. OPPD's proposed application of NEM in Cycle 14 was consistent with the same method used in the derivation of the biases and uncertainties of Reference 2.

References:

1. "The ROCS & DIT Computer Codes for Nuclear Design", CENPD-266-P-A, April 1983.

2. "Physics Biases and Uncertainties", CE-CES-129, Revision 1-P, August 1991.