Omaha Public Power District
Fort Calhoun Station
Unit No. 1

Cycle 14
Reload Evaluation

FORT CALHOUN STATION UNIT NO. 1
CYCLE 14
RELOAD EVALUATION

TABLE OF CONTENTS

FORT CALHOUN STATION UNIT NO. 1
CYCLE 14
RELOAD EVALUATION

TABLE OF CONTENTS (Continued)

1.0 INTRODUCTION AND SUMMARY

This report provides an evaluation of the design and performance for the operation of Fort Calhoun Station Unit No. 1 during its fourteenth fuel cycle at a full rated power of 1500 MWt. Planned operating conditions remain the same as those for Cycle 13, unless otherwise noted in the proposed Core Operating Limits Report and Technical Specification changes.

The core will consist of 80 presently operating Batches M, N and P assemblies, 52 fresh Batch R assemblies and 1 Batch M assembly discharged from a previous cycle.

The Cycle 14 analysis is based on a Cycle 13 termination point between 14,250 MWD/MTU and 15,250 MWD/MTU. In performing analyses of design basis events, limiting safety system settings and limiting conditions for operation, limiting values of key parameters were chosen to assure that expected Cycle 14 conditions would be enveloped, provided the Cycle 13 termination point falls within the above range. The analysis presented herein will accommodate a Cycle 14 length of up to 14,000 MWD/MTU with a coastdown of an additional 1,000 MWD/MTU.

The evaluation of the reload core characteristics has been conducted with respect to the Fort Calhoun Station Unit No. 1 Cycle 13 safety analysis described in the 1991 update of the USAR, hereafter referred to as the "reference cycle" in this report unless noted otherwise.

Specific core differences have been accounted for in the present analysis. In all cases, it has been concluded that either the reference cycle analyses envelope the new conditions or the revised analyses presented herein continue to show acceptable results. Where dictated by variations from the previous cycle, proposed modifications to the Technical Specifications have been provided or are being incorporated into the Cycle 14 Core Operating Limits Report.

The Cycle 14 core has been designed to minimize the neutron flux to limiting reactor pressure vessel welds to reduce the rate of RT_{PTS} shift on these welds. This will maximize the time to reach the screening criteria that is consistent with the procedure for calculating the amount of radiation embrittlement that a reactor vessel receives given in Regulatory Guide 1.99, Revision 2 and recently incorporated into 10 CFR 50.61.

The reload analysis presented in this report was performed utilizing the methodology documented in Omaha Public Power District's reload analysis methodology reports (References 1, 2, and 3).

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.

It is estimated that Cycle 13 will be terminated on or about February 1, 1992. The Cycle 13 termination point can vary between 14,250 MWD/MTU and 15,250 MWD/MTU and still be within the assumptions of the Cycle 14 analyses. As of November 3, 1991, the Cycle 13 burnup had reached 12,569 MWD/MTU.

3.0 GENERAL DESCRIPTION

The Cycle 14 core will consist of the number and type of assemblies and fuel batches shown in Table 3-1. Eight L assemblies, 41 M assemblies and 4 N assemblies will be discharged this outage. They will be replaced by 4 fresh Batch R1 assemblies (0.74 w/o natural enrichment), 16 fresh Batch R2 assemblies (3.85 w/o average enrichment with 28 IFBA rods at 0.003 gm B_{10} /inch), 4 fresh Batch R3 assemblies (3.85 w/o average enrichment with 48 IFBA rods at 0.003 gm B_{10} /inch), 8 fresh Batch R4 assemblies (3.85 w/o average enrichment with 64 IFBA rods at 0.003 gm B_{10} /inch), 12 fresh Batch R5 assemblies (3.85 w/o average enrichment with 84 IFBA rods at 0.003 gm B_{10} /inch), 4 fresh Batch R6 assemblies (3.60 w/o average enrichment with 84 IFBA rods at 0.003 gm B_{10} /inch) and 4 fresh Batch R7 assemblies (3.60 w/o average enrichment with 64 IFBA rods at 0.003 gm B_{10} /inch). In addition, the center assembly will be replaced by a Batch M assembly which was discharged after Cycle 12 and is currently residing in Region 1 of the spent fuel pool.

Figure 3-1 shows the fuel management pattern to be employed in Cycle 14. Several changes in fuel management strategy have been incorporated for Cycle 14. First, the overall fuel management scheme is designed to maximize the reduction in neutron leakage seen by the reactor vessel and limiting vessel weld locations. This strategy is called "extreme low radial leakage fuel management" and is very similar to the fuel management previously used in the Cycle 10 core loading pattern. Listed below are the specific changes which comprise the extreme low radial leakage fuel management strategy:

1) Twelve fuel assemblies on the core periphery will contain four full-length hafnium flux suppression rods per fuel assembly to locally reduce neutron flux near the limiting reactor vessel welds. Each of the hafnium rods will be placed in one of the outer CEA guide tubes of peripheral fuel assemblies.

2) Four fuel assemblies will contain natural uranium fuel rods which are located on the core periphery adjacent to the reactor vessel limiting welds. These four peripheral assembly locations could not support the use of full-length hafnium flux suppression rods due to the residence of CEA Shutdown Group A rods.

3) Use of an integral fuel burnable absorber (IFBA) instead of the traditional fuel displacing poison rods within selected new fuel assemblies. The IFBA rods consist of fuel pellets treated with an electrostatically applied, zirconium-diboride coating which surrounds the fuel pellet circumference. By using IFBA rods, extreme low radial leakage fuel management can provide greater reduction in vessel flux by increasing the number of fuel rods available to produce the rated power of 1500 MWt, thus gaining radial peaking factor margin which is needed to absorb the inward roll of the core power distribution caused, in part, by the peripheral flux reduction.

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, the average initial enrichment is 3.81 w/o U-235. For the second consecutive cycle, the fuel assembly zone loading technique is used to lower the radial power peaking factors within Batches R2 through R7. Batch R2 through R5 assemblies have fuel rods at both 4.0 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 fuel average discharge exposure at the end of Cycle 13 is projected to be 15,000 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 28,459 MWD/MTU.

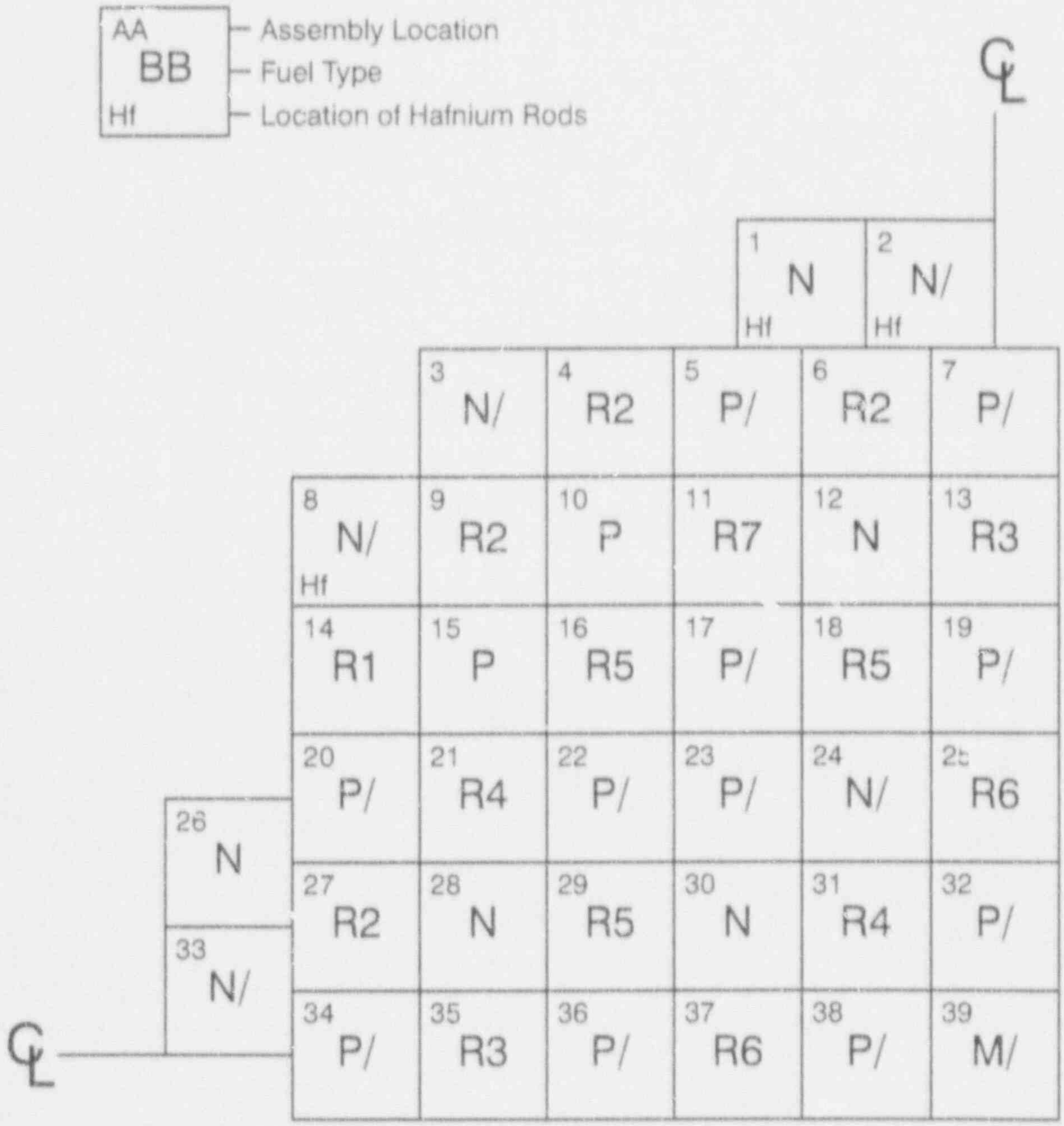
TABLE 3-1

FORT CALHOUN UNIT NO. 1
CYCLE 14 CORE LOADING

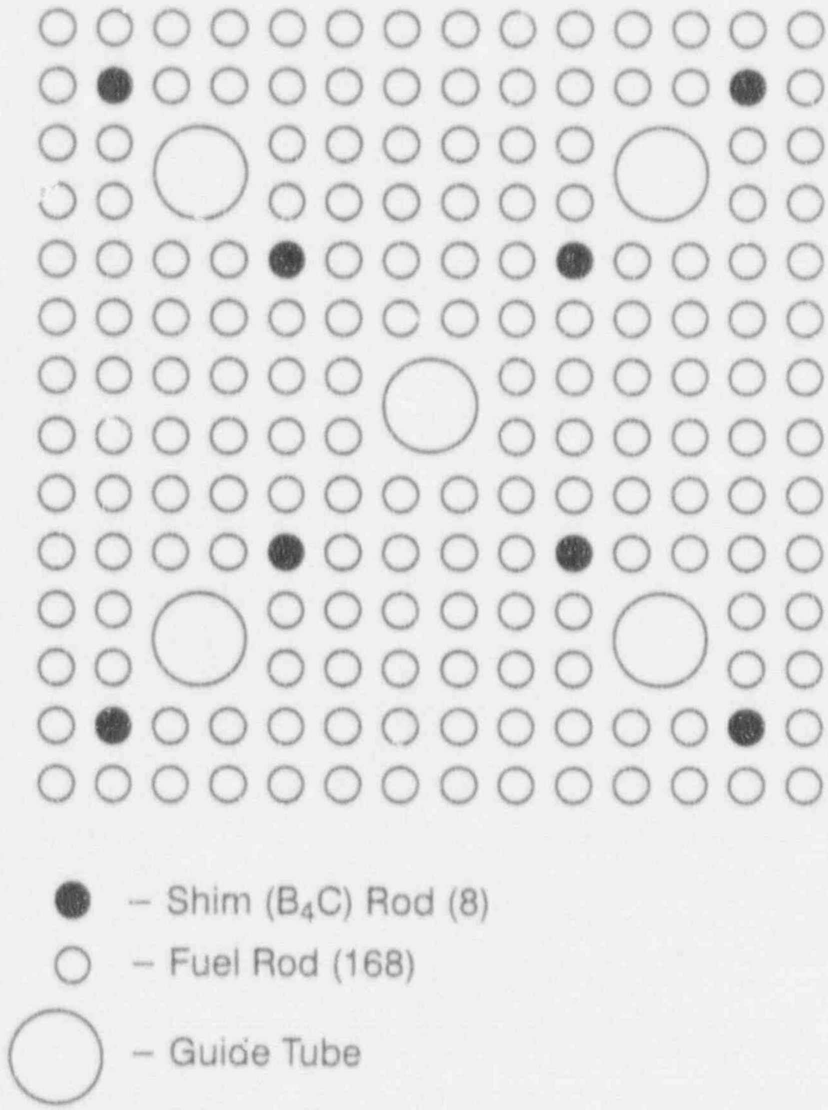
Assembly Designation	Number of Assemblies	BOC Avg. Burnup* MWD/MTU	EOC Avg. Burnup** MWD/MTU	Poison Rods per Assembly	IFBA Rods per Assembly	Initial Poison Loading gm B_{10}/in.
M/	1	30,957	45,607	8	–	0.024
N	20	28,485	38,842	0	–	0
N/	20	31,877	38,303	8	–	0.020
P	8	13,616	30,170	0	–	0
P/	32	19,256	34,392	8	–	0.027
R1	4	0	4,371	–	0	0.003
R2	16	0	13,996	–	28	0.003
R3	4	0	19,902	–	48	0.003
R4	8	0	19,704	–	64	0.003
R5	12	0	20,739	–	84	0.003
R6	4	0	20,419	–	84	0.003
R7	4	0	19,170	–	64	0.003

* Assumes EOC13=15,250 MWD/MTU

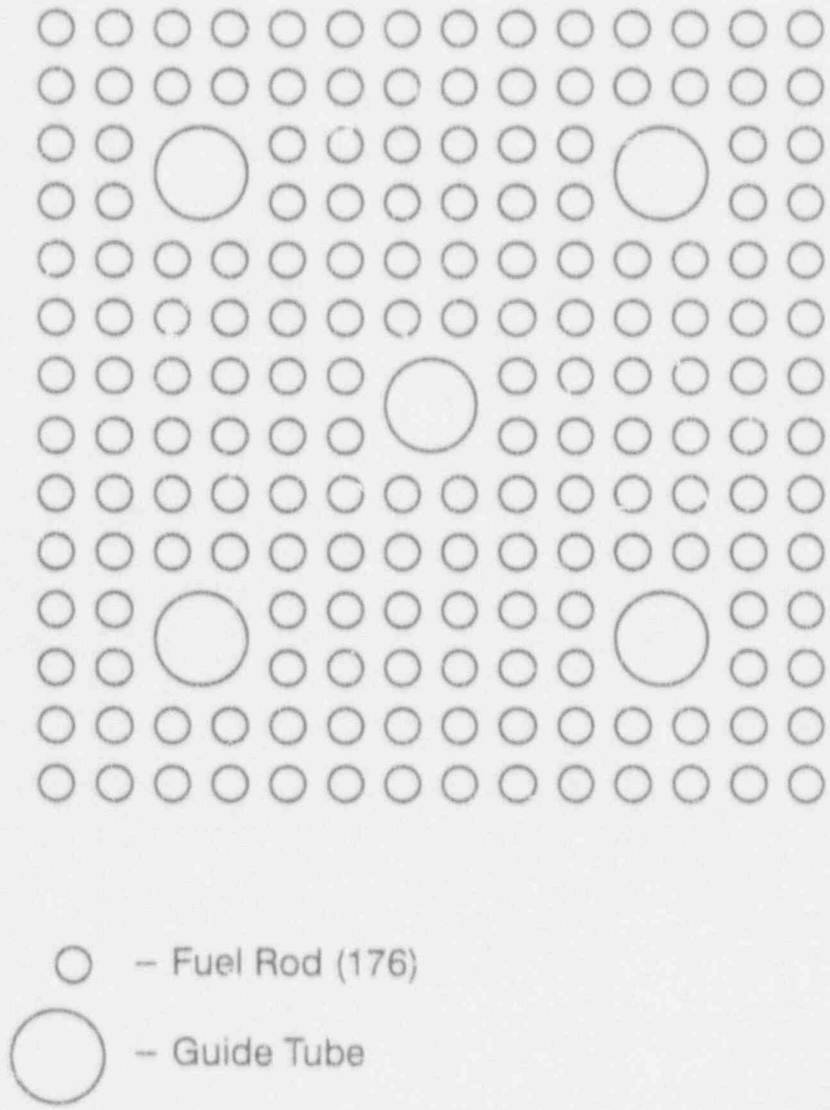
** Assumes EOC14=14,000 MWD/MTU



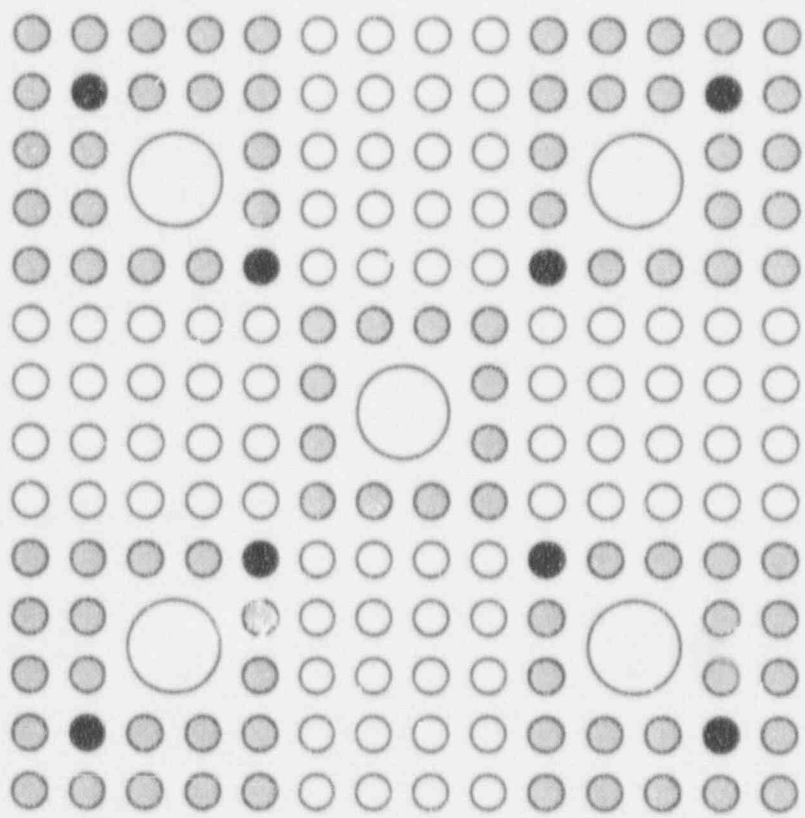
Cycle 14 Core Loading Pattern	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3-1



Batches M/ and N/ Assembly Fuel Rod and Poison Rod Locations	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3–2

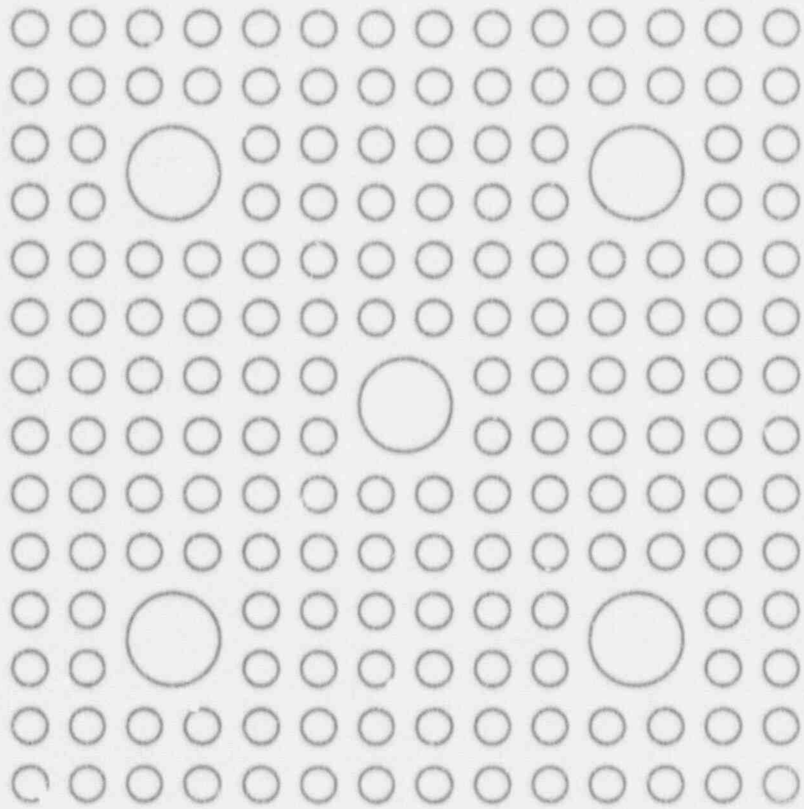


Batches N and P Fuel Rod Locations	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3-3



- Shim (B_4C) Rod (8)
- Low (3.25 $^w/_o$) Enrichment Fuel Rod (88)
- High (3.95 $^w/_o$) Enrichment Fuel Rod (80)
- Guide Tube

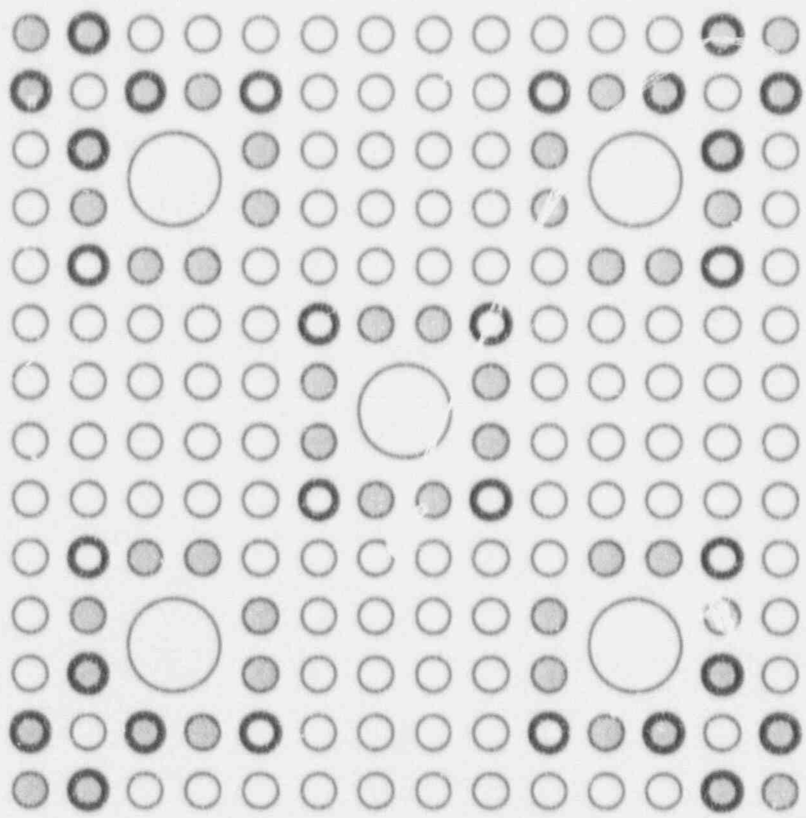
Batch P/ Assembly Fuel Rod and Poison Rod Locations	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3-4



○ – Natural Enriched Fuel Rod (176)

◯ – Guide Tube

Batch R1 Assembly Fuel Rod Locations	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3−5



– Low (3.5 $^{w}/_{o}$) Enrichment Fuel Rod (36)

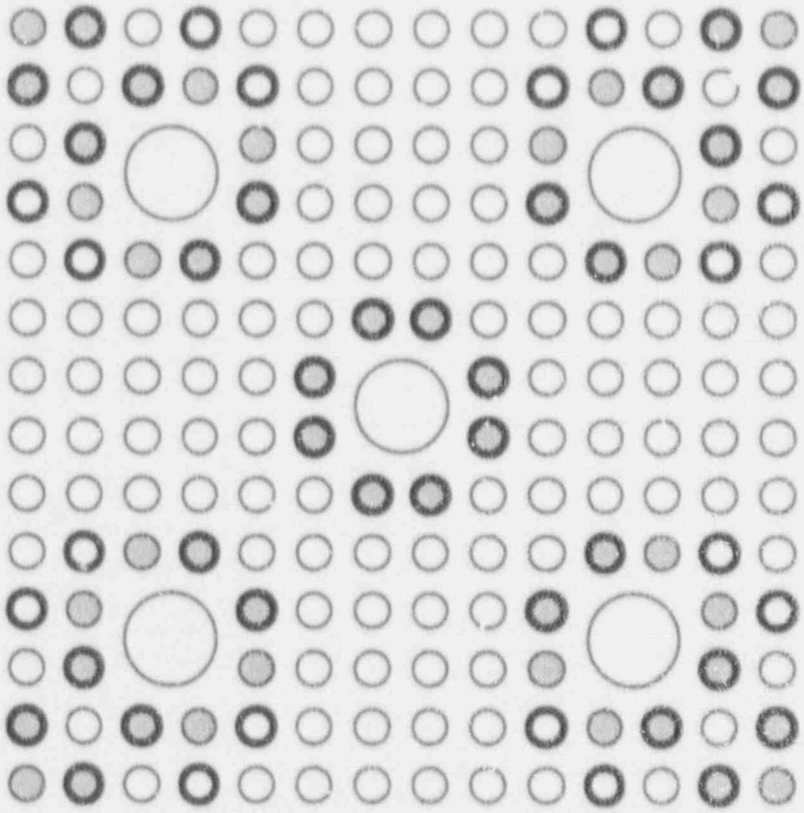
– Low (3.5 $^{w}/_{o}$) Enrichment Fuel Rod with IFBA (16)

– High (4.0 $^{w}/_{o}$) Enrichment Fuel Rod (112)

– High (4.0 $^{w}/_{o}$) Enrichment Fuel Rod with IFBA (12)

– Guide Tube

Batch R2 Assembly Fuel Rod and 28 IFBA Rod Locations	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3–6



– Low (3.5 $^{w}/_{o}$) Enrichment Fuel Rod (20)

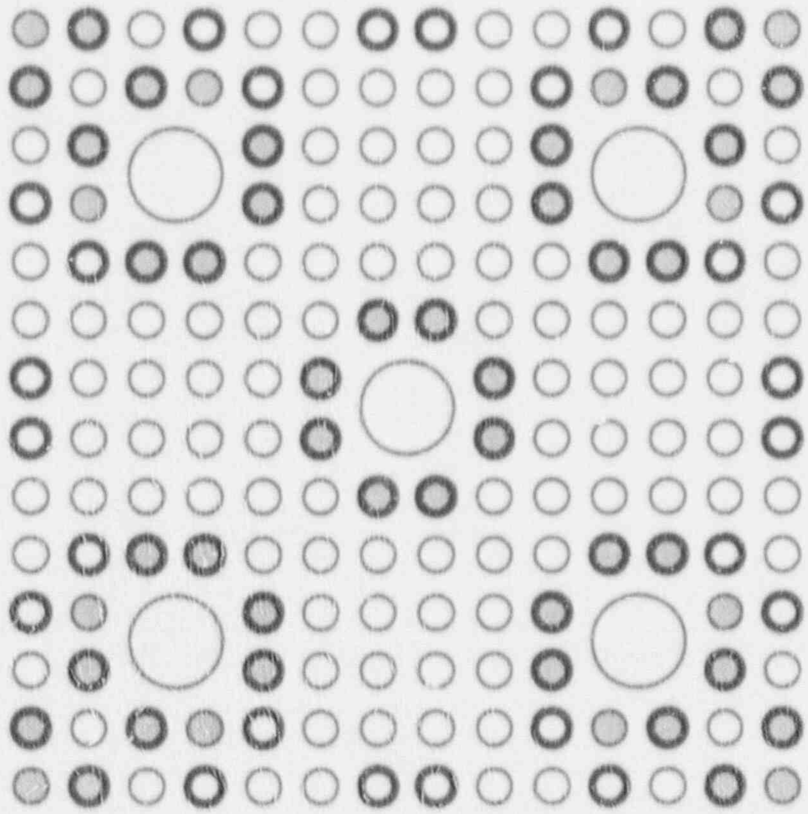
– Low (3.5 $^{w}/_{o}$) Enrichment Fuel Rod with IFBA (32)

– High (4.0 $^{w}/_{o}$) Enrichment Fuel Rod (108)

– High (4.0 $^{w}/_{o}$) Enrichment Fuel Rod with IFBA (16)

– Guide Tube

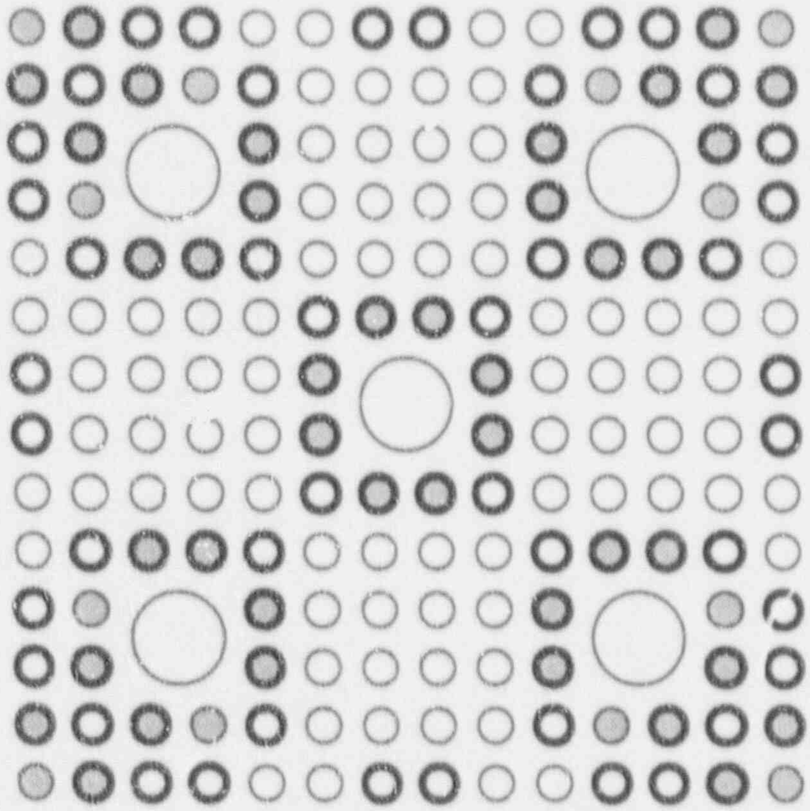
Batch R3 Assembly Fuel Rod and 48 IFBA Rod Locations	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3–7



- Low Enrichment Fuel Rod (12)
- Low Enrichment Fuel Rod with IFBA (40)
- High Enrichment Fuel Rod (100)
- High Enrichment Fuel Rod with IFBA (24)
- Guide Tube

Batch	Low Enrichment (w/o)	High Enrichment (w/o)
R4	3.5	4.0
R7	3.25	3.75

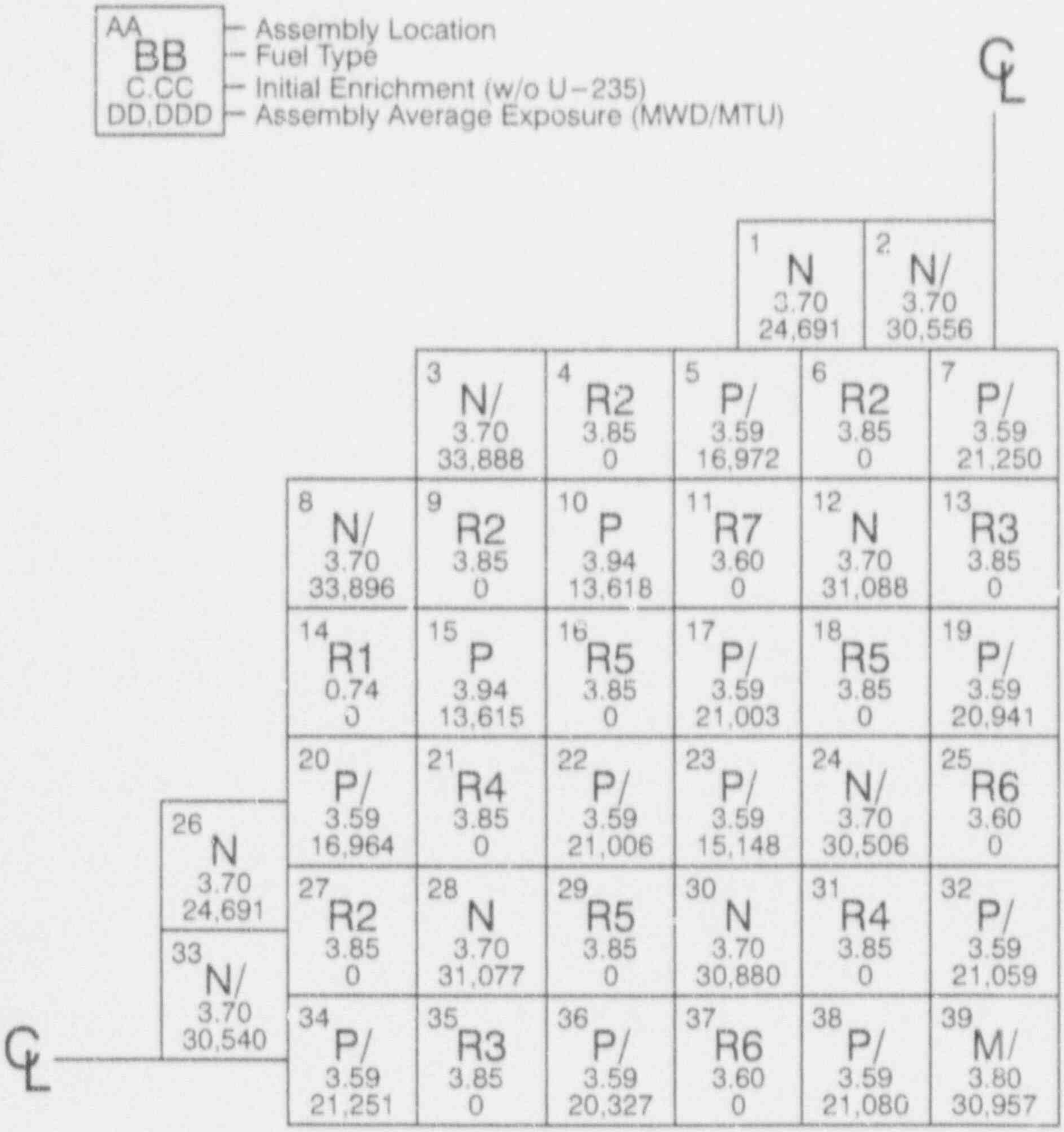
Batches R4 and R7 Assembly Fuel Rod and 64 IFBA Rod Locations	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3-8



- Low Enrichment Fuel Rod (12)
- Low Enrichment Fuel Rod with IFBA (40)
- High Enrichment Fuel Rod (80)
- High Enrichment Fuel Rod with IFBA (44)
- Guide Tube

Batch	Low Enrichment (w/o)	High Enrichment (w/o)
R5	3.5	4.0
R6	3.25	3.75

Batches R5 and R6 Assembly Fuel Rod and 84 IFBA Rod Locations	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3-9



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

AA	Assembly Location
BB	Fuel Type
[illegible].CC	Initial Enrichment (w/o U-235)
DD,DDD	Assembly Average Exposure (MWD/MTU)

℄

Assembly Location	Fuel Type	Initial Enrichment	Assembly Average Exposure
1	N	3.70	28,583
2	N/	3.70	33,931
3	N/	3.70	38,924
4	R2	3.85	11,964
5	P/	3.59	28,658
6	R2	3.85	14,692
7	P/	3.59	32,892
8	N/	3.70	36,825
9	R2	3.85	13,972
10	P	3.94	30,899
11	R7	3.60	19,170
12	N	3.70	45,196
13	R3	3.85	19,719
14	R1	0.74	4,37[illegible]
15	P	[illegible]	[illegible]9,441
16	R5	3.85	20,891
17	P/	3.59	38,317
18	R5	3.85	20,673
19	P/	3.59	38,469
20	P/	3.59	27,565
21	R4	3.85	18,788
22	P/	3.59	36,166
23	P/	3.59	32,754
24	N/	3.70	45,878
25	R6	3.60	20,383
26	N	3.70	29,286
27	R2	3.85	15,354
2[illegible]	[illegible]	3.70	45,250
29	R5	3.85	20,652
30	N	3.70	45,896
31	R4	3.85	20,620
32	P/	3.59	37,929
33	N/	3.70	35,955
34	P/	3.59	33,981
35	R3	3.85	20,084
36	P/	3.59	38,069
37	R6	3.60	20,456
38	P/	3.59	38,007
39	M/	3.80	45,607

℄

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

4.0 FUEL SYSTEMS DESIGN

The mechanical design for the Batch R fuel is slightly different from the Batch P fuel due to a change to Westinghouse (W) as the fuel vendor.

The Batch R fuel is similar in design to the fuel supplied by Combustion Engineering and is mechanically, thermally, and hydraulically compatible with the ABB–CE fuel remaining in the Cycle 14 core. References 4 and 5 describe Batches M and P fuel characteristics and design, respectively. The Westinghouse fuel will not be resident in the reactor with any of the Exxon (Siemens) fuel previously used at Fort Calhoun.

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 between 14,250 MWD/MTU and 15,250 MWD/MTU and limiting values established for the safety analysis. The Cycle 14 loading pattern is valid for any Cycle 13 end[illegible] between these values.

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 Main Steam Line Break accident is the most limiting accident of those used in the determination of required shutdown margin

5.0 NUCLEAR DESIGN (Continued)

5.1 PHYSICAL CHARACTERISTICS (Continued)

5.1.1 Fuel Management (Continued)

for compliance with Technical Specifications. Table 5–2 presents a summary of CEA shutdown worths and reactivity allowances for the Cycle 14 BOC, HZP, MSLB accident. The Cycle 14 values, calculated for 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, based on an EOC13 burnup of 15,250 MWD/MTU.

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 2.095, including uncertainty allowances.

5.0 NUCLEAR DESIGN (Continued)

5.1 PHYSICAL CHARACTERISTICS (Continued)

5.1.3 Safety Related Data

5.1.3.1 Ejected CEA Data

Bounding reactivity worth and planar power peaking factors associated with an ejected CEA event are shown in Table 5–3 for both the beginning and end of Cycle 14. These values are projected to encompass the worst conditions anticipated during Cycles 14 through 16. The values shown bound actual Cycle 14 values which were calculated in accordance with Reference 3. In addition, Table 5–3 lists these values used from Cycle 13 for comparison.

5.1.3.2 Dropped CEA Data

The Cycle 14 safety related data for the dropped CEA analysis were calculated identically with the methods used in Cycle 13.

5.2 ANALYTICAL INPUT TO INCORE MEASUREMENTS

Incore detector measurement constants to be used in evaluating the reload cycle power distributions will be calculated in the same manner as for Cycle 13.

5.3 NUCLEAR DESIGN METHODOLOGY

Analyses have been performed in the manner and with the methodologies documented in References 1 and 2.

5.4 UNCERTAINTIES IN MEASURED POWER DISTRIBUTIONS

The power distribution measurement uncertainties which are applied to Cycle 14 are the same as those presented in Reference 2.

TABLE 5-1

FORT CALHOUN UNIT NO. 1, CYCLE 14
NOMINAL PHYSICS CHARACTERISTICS

	Units	Cycle 13	Cycle 14
Critical Boron Concentration			
Hot Full Power, ARO, Equilibrium Xenon, BOC	ppm	1187	835
Inverse Boron Worth			
Hot Full Power, BOC	ppm/%$\Delta\rho$	112	113
Hot Full Power, EOC	ppm/%$\Delta\rho$	84	90
Reactivity Coefficients with All CEAs Withdrawn			
Moderator Temperature Coefficient (MTC) (Includes uncertainties)			
Beginning of Cycle, HZP	$10^{-4}\Delta\rho/°F$	+0.51*	+0.09
End of Cycle, HFP	$10^{-4}\Delta\rho/°F$	-2.47	-2.80**
Doppler Coefficient (FTC)			
Hot Full Power, BOC	$10^{-5}\Delta\rho/°F$	-1.66	-1.51
Hot Full Power, EOC	$10^{-5}\Delta\rho/°F$	-1.85	-1.69
Total Delayed Neutron Fraction, β_{eff}			
BOC		0.00614	0.00625
EOC		0.00519	0.00518
Neutron Generation Time, l*			
BOC	10^{-6}sec	21.6	21.6
EOC	10^{-6}sec	28.8	27.2

* This value exceeds the Technical Specification limit of +0.50 x $10^{-4}\Delta\rho/°F$, however, the actual MTC at HZP, BOC, including uncertainties, did not exceed the Technical Specification limit.

** This value exceeds the current Technical Specification limit of -2.70 x $10^{-4}\Delta\rho/°F$, therefore, a change to Technical Specification 2.10.2(3)c. is being made to lower the limit to -3.00 x $10^{-4}\Delta\rho/°F$ including uncertainties.

TABLE 5-2

FORT CALHOUN UNIT NO. 1, CYCLE 14
LIMITING VALUES OF REACTIVITY WORTHS AND ALLOWANCES
FOR HOT ZERO POWER
MAIN STEAM LINE BREAK, %$\Delta\rho$

	Cycle 13	Cycle 14
1. Worth of all CEAs Inserted	9.23	7.52
2. Stuck CEA Allowance	1.83	1.17
3. Worth of all CEAs Less Worth of Most Reactive CEA Stuck Out	7.40	6.35
4. Power Dependent Insertion Limit CEA Worth	1.23	1.19
5. Calculated Scram Worth	6.17	5.16
6. Physics Uncertainty plus Bias	0.80*	0.10**
7. Net Available Scram Worth	5.37	5.06
8. Technical Specification Shutdown Margin	4.00	4.00
9. Margin in Excess of Technical Specification Shutdown Margin	1.37	1.06

* 13% of calculated scram worth.

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

TABLE 5-3

FORT CALHOUN UNIT NO. 1, CYCLE 14
BOUNDING CEA EJECTION DATA

Maximum Radial Power Peaking Factor	BOC 13 Value	EOC 13 Value	BOC 14 Value	EOC 14 Value
Full Power with Bank 4 inserted; worst CEA ejected	2.41	2.68	3.73	3.73
Zero Power with Banks 4+3 inserted; worst CEA ejected	3.43	3.99	5.74	5.74
Maximum Ejected CEA Worth (%$\Delta\rho$)				
Full Power with Bank 4 inserted worst CEA ejected	0.22	0.30	0.36	0.36
Zero Power with Banks 4+3 inserted worst CEA ejected	0.28	0.48	0.69	0.69

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

					1 0.2751	2 0.2347
		3 0.372[illegible]	4 0.9116	5 0.8667	6 1.0990	7 0.8673
	8 0.2018	9 1.0680	10 1.3302	11 1.3466	12 1.0139	13 1.4143
	14 0.2459	15 1.1808	16 1.4544 1.6285	17 1.2804	18 1.3796	19 1.2332
	20 0.7583	21 1.2985	22 1.2603	23 1.3288	24 1.0904	25 1.3210
26 0.3264	27 1.1522	28 1.0163	29 1.3734	30 1.0589	31 1.4306	32 1.2305
33 0.3937	34 0.9643	35 1.4475	36 1.2481	37 1.3249	38 1.2349	39 1.1051

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

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

					1 0.2661	2 0.2308
		3 0.3440	4 0.8433	5 0.8142	6 1.0429	7 0.8104
	8 0.2007	9 0.9873	10 1.2064	11 1.3964	12 0.9951	13 1.4327
	14 0.3106	15 1.1085	16 1.5300	17 1.2296	18 1.5334	19 1.2628
26 0.3123	20 0.7390	21 1.3692	22 1.2195	23 1.2450	24 1.0973	25 1.5255
33 0.3670	27 1.0861	28 0.9987	29 1.5319 1.7370	30 1.0716	31 1.5193	32 1.2054
	34 0.8815	35 1.4567	36 1.2780	37 1.5311	38 1.2095	39 1.0309

℄ ℄

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

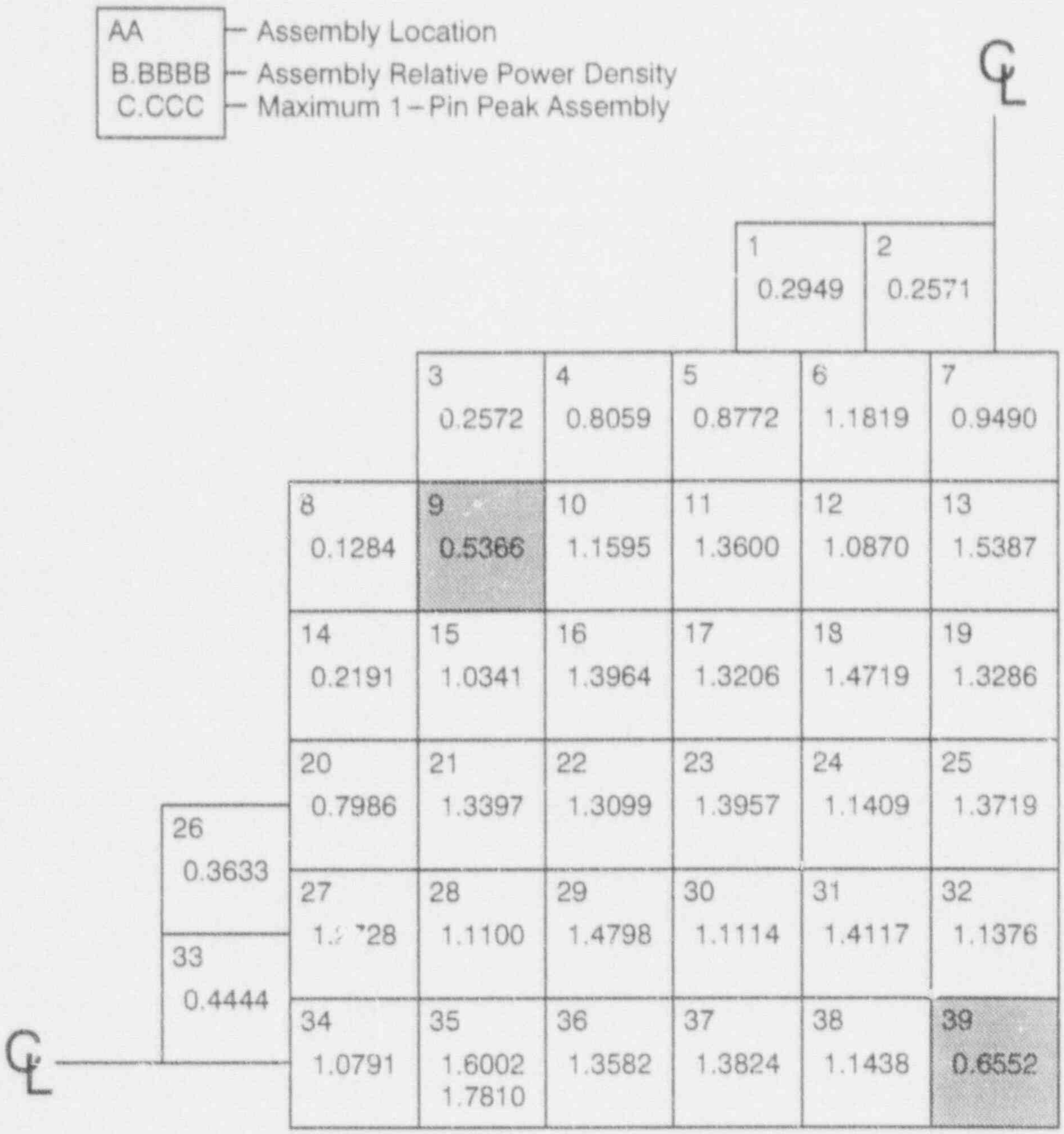
AA	Assembly Location
B.BBBB	Assembly Relative Power Density
C.CCC	Maximum 1–Pin Peak Assembly

℄

					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
26 0.3851	20 0.8223	21 1.3852	22 1.1702	23 1.1710	24 1.0403	25 1.4233
33 0.4494	27 1.1709	28 1.0130	29 1.4674	30 1.0194	31 1.4047	32 1.1143
℄	34 0.9477	35 1.4310	36 1.2082	37 1.4289	38 1.1175	39 0.9687

Maximum 1–Pin Peak at 17% Core Height

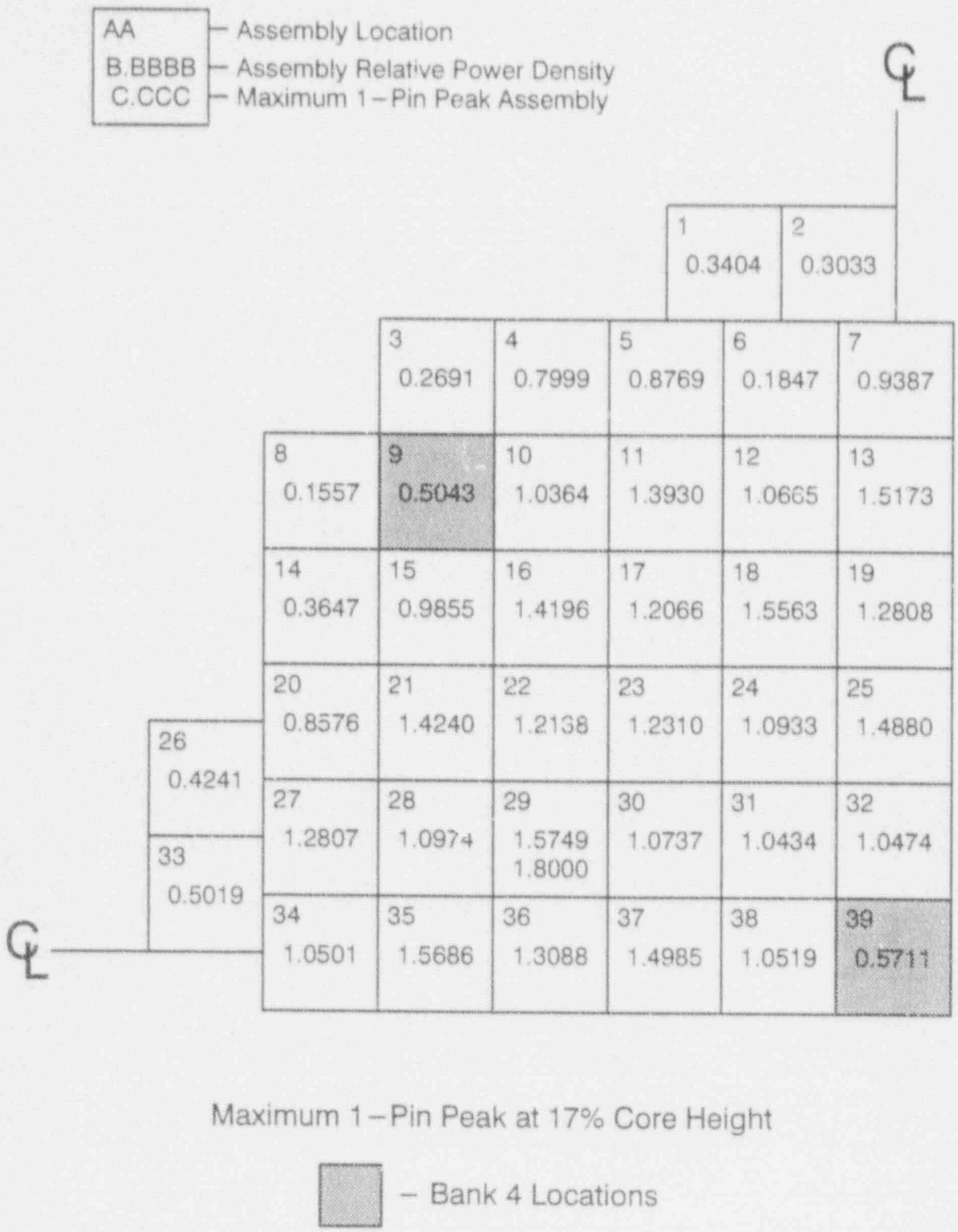
Cycle 14 Assembly Power Distribution 14,000 MWD/MTU, HFP, Eq. Xenon	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 5–3



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



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. This is different from the combination that was used in the Cycle 8 through Cycle 13 Fort Calhoun reload analyses (References 4 through 9) in that the more accurate TORC code was used in place of the CETOP–D code. 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 Westinghouse fuel does not have any DNBR penalty associated with the design requirements for the Westinghouse fuel based on NRC fuel bowing requirements, thus, the more limiting CE fuel bow penalty was used in the analyses.

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	°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.

7.0 TRANSIENT ANALYSIS

This section presents the results of the Omaha Public Power District Fort Calhoun Station Unit 1, Cycle 14 Non-LOCA safety analyses at 1500 MWt.

The Design Bases Events (DBEs) considered in the safety analysis are listed in Table 7-1. These events were categorized in the following groups:

1. Anticipated Operational Occurrences (AOOs) for which the intervention of the Reactor Protection System (RPS) is necessary to prevent exceeding acceptable limits.
2. AOOs for which the initial steady state thermal margin, maintained by Limiting Conditions for Operation (LCO), are necessary to prevent exceeding acceptable limits.
3. Postulated Accidents.

Core parameters input to the safety analyses for evaluating approaches to DNB and centerline temperature to melt fuel design limits are presented in Table 7-2.

As indicated in Table 7-1, no reanalysis was performed for the DBEs for which key transient input parameters are within the bounds (i.e., conservative with respect to) of the reference cycle values (Fort Calhoun Updated Safety Analysis Report including Cycle 13 analyses, Reference 1). For these DBEs the results and conclusions quoted in the reference cycle analysis remain valid for Cycle 14.

For those analyses indicated as reviewed, calculations were performed in accordance with Reference 6 until a 10 CFR 50.59 determination could be made that Cycle 14 results would be bounded by Cycle 13 or the USAR reference cycle.

Events were evaluated for up to a total of 6% steam generator tube plugging in Cycle 11 where conservative. Fort Calhoun Station currently has 1.08% steam generator tubes plugged; thus, no additional analysis is required.

For the events reanalyzed, Table 7-3 shows the reason for the reanalysis, the acceptance criterion to be used in judging the results and a summary of the results obtained. Detailed presentations of the results of the reanalyses are provided in Sections 7.1 through 7.3.

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	Reviewed[3,5]
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*	1.78*
Bank 4 Inserted		1.73*	1.91*
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	$10^{-4}\ \Delta\rho/^{\circ}F$	-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 7-2
(Continued)

Safety Parameters	Units	Cycle 13 Values	Cycle 14 Values
Power Level	MWt	1500*	1500*
Maximum Steady State Temperature	°F	543*	545*
Minimum Steady State Pressurizer Pressure	psia	2075*	2075*
Maximum Augmentation Factor		1.000	1.000
Reactor Coolant Flow	gpm	202,500*	202,500*
Steam Generator Tube Plugging	%	6	6
Negative Axial Shape LCO Extreme Assumed at Full Power (Ex-Cores)	asiu	-0.18	-0.18
Maximum CEA Insertion at Full Power	% Insertion of Bank 4	25	25
Maximum Initial Linear Heat Rate for Transient Other than LOCA	kW/ft	14.4	13.8
Steady State Linear Heat Rate for Fuel CTM Assumed in the Safety Analysis	kW/ft	22.0	22.0
CEA Drop Time to 100% Including Holding Coil Delay	sec	3.1	3.1
Minimum DNBR (CE-1)		1.18*	1.18*

* The effects of uncertainties on these parameters were accounted for statistically in the DNBR and CTM calculations. The DNBR analysis 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 =1.72 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

7.0 TRANSIENT ANALYSIS (Continued)

7.1 ANTICIPATED OPERATIONAL OCCURRENCES (CATEGORY 1)

7.1.1 RCS Depressurization Event

The RCS Depressurization event was reanalyzed for Cycle 14 to determine the pressure bias term for the TM/LP trip setpoint.

The RCS Depressurization event is one of the Design Basis Events analyzed to determine the maximum pressure bias term input to the TM/LP trip. The methodology used for Cycle 14 is described in References 6 and 7. The pressure bias term accounts for margin degradation attributable to measurement and trip system processing delay times. Changes in core power, inlet temperature and RCS pressure during the transient are monitored by the TM/LP trip directly. Consequently, with TM/LP trip setpoints and the bias term determined in this analysis, adequate protection will be provided for the RCS Depressurization event to prevent the acceptable DNBR design limit from being exceeded. Table 7.1.1-1 provides a sequence of events for the RCS Depressurization analysis.

The analysis of this event shows that incorporating a pressure bias term of 30 psia in the TM/LP trip setpoints will ensure that the RPS provides adequate protection to prevent the acceptable DNBR design limit from being exceeded during an RCS Depressurization event.

The RCS Depressurization event is the only event that is currently analyzed to determine the pressure bias term, since the Excess Load event has been reclassified as an event requiring initial margin for protection. The Excess Load event is discussed in section 7.2.1.

TABLE 7.1.1-1

FORT CALHOUN UNIT NO.1, CYCLE 14
SEQUENCE OF EVENTS FOR RCS DEPRESSURIZATION

Time (sec)	Event	Setpoint or Value
0.000	Inadvertent Opening of Both Pressurizer Power Operated Relief Valves	------
7.382	Reactor Trip	2075.75 psia
9.409	Time of Minimum DNBR	2047.16 psia

7.0 TRANSIENT ANALYSIS (Continued)

7.2 ANTICIPATED OPERATIONAL OCCURRENCES (CATEGORY 2)

7.2.1 Excess Load Event

The Excess Load event was reclassified for Cycle 14 from an event which is protected by an RPS trip to an event which is protected by the RPS and sufficient initial thermal margin which is maintained by the LCOs. This reclassification does not result in a net gain in margin. It only transfers the margin requirements from the LSSS to the LCO.

The Excess Load event was analyzed for Cycle 14 to determine the DNB and LHR ROPMs which are used to ensure sufficient margin is included in the DNB and LHR LCOs to provide protection to the fuel design limits in the event of an Excess Load event. The methodology used to perform the analysis is described in Reference 6. The key input parameters used in the Cycle 14 Excess Load analysis are presented in Table 7.2.1-1.

It is assumed in the analysis that the reactor will trip on Variable High Power during an excess load event. Therefore, the key to the analysis is maximizing the time between the initiation of the event (instantaneous opening of the steam dump and bypass valves) and the time at which the Variable High Power trip (VHPT) signal is generated. Several assumptions are made to maximize this time. Since the VHPT uses the auctioneered higher value of the excore power signal and ΔT-Power calculator, an MTC is chosen which ensures that the ΔT-Power calculator and the excore detectors both reach the VHPT setpoint at the same time. The maximum temperature shadowing factor is used to maximize the decalibration of the excore detectors due to RCS cooldown. Also, the time constants for the hot and cold leg resistance temperature detectors (RTDs) are chosen to maximize the lag between the ΔT-Power calculator and the actual core heat flux.

The DNB and LHR ROPMs calculated for the Excess Load event are compared to those calculated for other AOO events such as the CEA Drop and CEA Withdrawal to determine the most conservative (largest) ROPMs to input to the calculation of the LCOs. This ensures that there will be sufficient margin included in the LCOs to protect all AOO events requiring initial margin for protection.

It was concluded from the Cycle 14 analysis that the ROPM required by the Excess Load event was bounded by the requirements of the CEA Drop Event.

TABLE 7.2.1-1

FORT CALHOUN UNIT NO. 1, CYCLE 14
KEY PARAMETERS ASSUMED IN THE EXCESS LOAD ANALYSIS

Parameter	Units	Cycle 14
Initial Core Power Level	MWt	1530
Core Inlet Coolant Temperature	°F	547
Pressurizer Pressure	psia	2053
Moderator Temperature Coefficient	$\times 10^{-4}\ \Delta\rho/°F$	−0.707
Doppler Coefficient Multiplier		0.85
CEA Worth at Trip	$\%\Delta\rho$	5.7922
Excore Temperature Shadowing Factor	%/°F	0.35
Cold Leg RTD Time Constant	sec	12.0
Hot Leg RTD Time Constant	sec	3.0

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 29.1% (19.1% 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 5.46 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.

7.0 TRANSIENT ANALYSIS (Continued)

7.2 ANTICIPATED OPERATIONAL OCCURRENCES (CATEGORY 2) (Continued)

7.2.2 CEA Withdrawal Event (Continued)

It may be concluded that the CEA Withdrawal event, when initiated from the Technical Specification LCOs (in conjunction with the Variable High Power Trip, if required), will not lead to a DNBR or fuel temperature which violates the DNBR and CTM design limits. It was further concluded that the initial available overpower margin requirements for this event were bounded by that of the CEA Drop event.

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	6.407
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 reviewed for Cycle 14 and it was determined that the event was bounded by the reference cycle (Cycle 12) analysis. The input parameters are listed for Cycles 12 and 14 for comparison in Table 7.2.3−1.

Thus, it was concluded that the reference cycle analysis is bounding for Cycle 14 operation.

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	93
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	1.78

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

7.0 TRANSIENT ANALYSIS (Continued)

7.2 ANTICIPATED OPERATIONAL OCCURRENCES (CATEGORY 2) (Continued)

7.2.4 Full Length CEA Drop Event

The Full Length CEA Drop 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 CTM design limits will not be exceeded.

This event was analyzed parametrically in initial axial shape and rod configuration using the methods described in Reference 6. Table 7.2.4-1 lists the key input parameters used for Cycle 14 and compares them to the reference cycle (Cycle 11) values while Table 7.2.4-2 contains a sequence of events for the CEA Drop analysis.

The transient was conservatively analyzed at full power with an ASI of −0.182, which is outside of the LCO limit of −0.06. This results in a minimum CE−1 DNBR of 1.377. A maximum allowable initial linear heat generation rate of 18.4 kW/ft could exist as an initial condition without exceeding the acceptable fuel CTM limit of 22 kW/ft during this transient. This amount of margin is assured by setting the LHR related LCOs based on the more limiting allowable LOCA linear heat rate.

It can be concluded that the CEA Drop event was the most limiting of the AOO's dependent upon initial available overpower margin. When initiated from the Technical Specification LCOs, the event will not exceed the DNBR CTM design limits.

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, %$\Delta\rho$	−0.2337	−0.2947
	PDIL, %$\Delta\rho$	−0.2295	−0.2940
Maximum Allowed Power Shape Index at Negative Extreme of LCO Band		−0.18	−0.18
Radial Peaking Distortion Factor	Unrodded Region	1.1566	1.1937
	Bank 4 Inserted	1.1598	1.1904

* 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	63.8% of 1500 MWt
78.97	Core Inlet Temperature Reaches a Minimum Value	538.68°F
199.9	RCS Pressure Reaches a Minimum Value	1996.10 psia
200.0	Core Power Returns to its Maximum Value	94.85% of 1500 MWt
200.0	Minimum DNBR is Reached	1.377 (CE-1 Correlation)

7.0 TRANSIENT ANALYSIS (Continued)

7.2 ANTICIPATED OPERATIONAL OCCURRENCES (CATEGORY 2) (Continued)

7.2.5 Boron Dilution Event

The Boron Dilution event was reviewed for Cycle 14 to verify that sufficient time is available for an operator to identify the cause and to terminate a boron dilution event for any mode of operation before SAFDL limits are violated.

Table 7.2.5-1 compares the values of the key transient parameters assumed in each mode of operation for Cycle 14 and the reference cycle, Cycle 13. The Cycle 14 analysis utilized a mass basis in the calculations, as was used in Cycle 13, rather than a volumetric basis to ensure that all operating temperature ranges for all modes of operation were bounded.

Since the critical boron concentration for Cycle 14 is less than the corresponding Cycle 13 values for all modes there is no further analysis required as the Cycle 13 results will bound Cycle 14.

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/%$\Delta\rho$		
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, %$\Delta\rho$		
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	1900

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

** Includes a 5.0%$\Delta\rho$ shutdown margin.

7.0 TRANSIENT ANALYSIS (Continued)

7.3 POSTULATED ACCIDENTS

7.3.1 CEA Ejection

The CEA Ejection event was reanalyzed for Cycle 14 since Westinghouse will be providing a new fuel design. A summary report was transmitted to the NRC for review in Reference 14.

7.3.2 Steam Line Break Accident

This accident was reviewed for Cycle 14 using the methodology discussed in References 6 and 12. The Steam Line Break (SLB) accident was previously analyzed in the Fort Calhoun FSAR and satisfactory results were reported therein. The SLB accidents at both HZP and HFP were examined in the reference cycle (Cycle 8) safety evaluation with acceptable results obtained. Both the FSAR and reference cycle evaluations are reported in the 1991 update of the Fort Calhoun Station Unit No. 1 USAR.

The Full Power Steam Line Break accident was reviewed for Cycle 14 for a more negative MTC of -3.0×10^{-4} $\Delta\rho/^{\circ}F$ than the -2.5×10^{-4} $\Delta\rho/^{\circ}F$ value that was used in the Cycle 8 analysis. However, the cooldown curve for Cycle 14 is bounded by Cycle 8 (as shown in Figure 7.3.2−1). This figure shows that the reactivity insertion for the Cycle 14 core with an MTC of -3.0×10^{-4} $\Delta\rho/^{\circ}F$ due to a SLB accident at full power is substantially less than the value used in the Cycle 8 analysis. (This smaller reactivity insertion is due to the use of the DIT cross−sections which are valid for a range of moderator temperatures from room temperature to 600°K while the analyses prior to Cycle 9 were performed with cooldown curves derived by conservatively extrapolating CEPAK cross−section values to low temperatures.) The Cycle 14 minimum available shutdown worth at HFP is 6.2885 %$\Delta\rho$ compared to a Cycle 8 value of 6.68%$\Delta\rho$. This implies a margin decrease of 0.395%$\Delta\rho$. The Cycle 14 moderator cooldown reactivity between 574°F and 350°F at HFP is 4.7%$\Delta\rho$ compared to 5.37 %$\Delta\rho$ in Cycle 8. This implies a margin increase of 0.67 %$\Delta\rho$. The Cycle 14 doppler coefficient is more negative than the Cycle 8 doppler including uncertainties. However, this loss in margin is offset by the gain in margin from the moderator cooldown reactivity. The net gain ensures that the overall reactivity insertion for a Cycle 14 SLB is less than that of the reference cycle analysis. Therefore, the return to power is less than that of the reference cycle and Cycle 1 FSAR analyses.

7.0 TRANSIENT ANALYSIS (Continued)

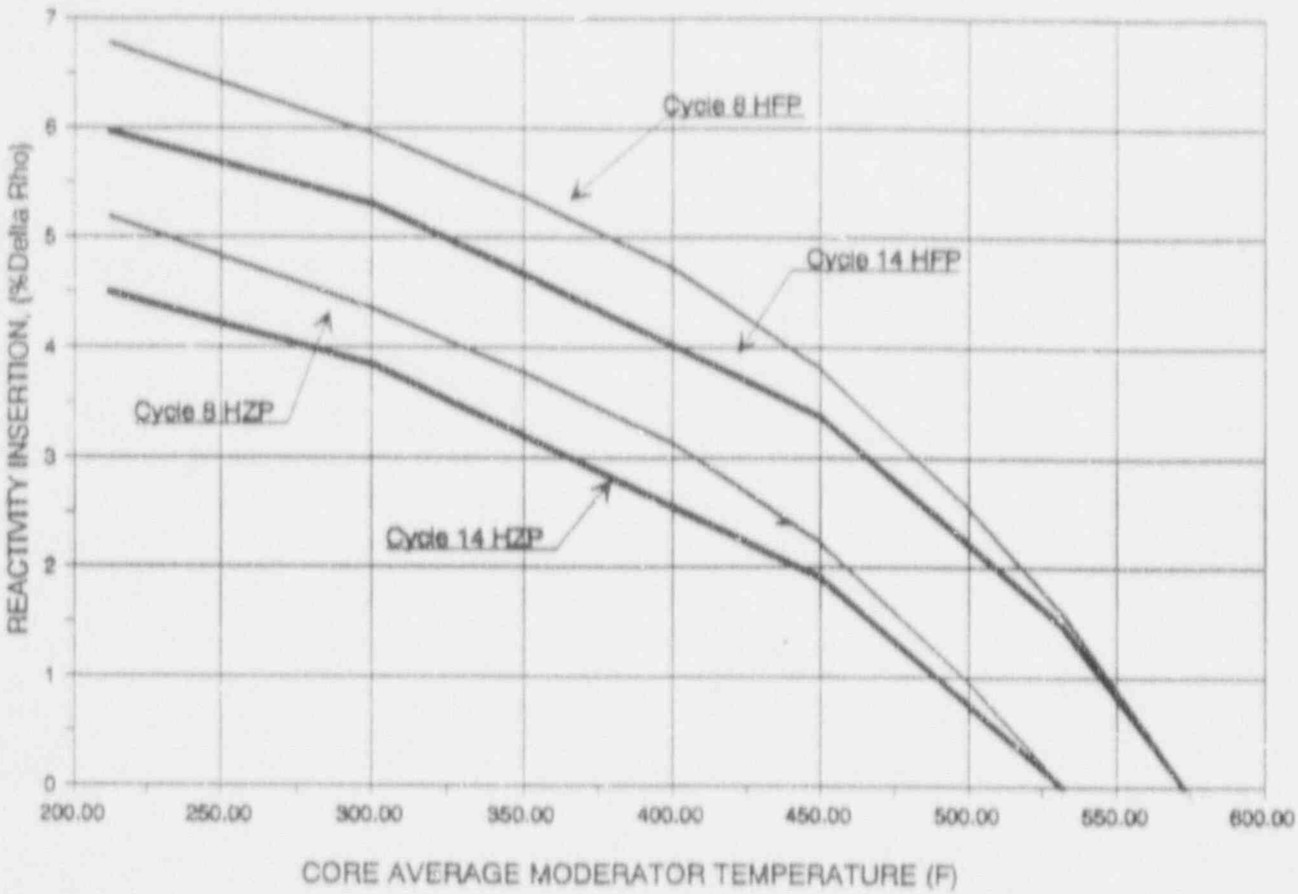
7.3 POSTULATED ACCIDENTS (Continued)

7.3.2 Steam Line Break Accident (Continued)

A similar evaluation was performed for the Hot Zero Power SLB accident. Again, the Cycle 14 cooldown for an MTC of -3.0×10^{-4} $\Delta\rho/^{\circ}F$ shows a substantially smaller reactivity insertion than was used in the Cycle 8 analysis (as seen in Figure 7.3.2-1). Since the minimum available shutdown margin for Cycle 14 remains unchanged from the reference cycle value (4.0%$\Delta\rho$), the overall reactivity insertion for the Cycle 14 SLB accident will be less severe than that reported for the reference cycle and the FSAR (Cycle 1) cases.

Based on the evaluation presented above, it is concluded that the consequences of a SLB accident initiated at either zero or full power are less severe than the reference cycle and FSAR (Cycle 1) cases.

Since a negative determination utilizing the 10 CFR 50.59 criteria was made for the Cycle 14 SLB accident, no reanalysis was performed. Thus, it was concluded that the reference cycle analysis is bounding for Cycle 14 operation.



Steam Line Break Accident Reactivity vs. Moderator Temperature	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 7.3.2−1

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 1.78 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.

8.0 ECCS PERFORMANCE ANALYSIS

Both Cycle 14 Large and Small Break Loss of Coolant accident analyses were performed by Westinghouse using the methodology discussed in Reference 1. A summary containing the results of the analyses was submitted in Reference 2. The peak linear heat generation rate of 15.5 kW/ft was conservatively reduced to 13.8 kW/ft for the non–LOCA transients to ensure the CE fuel mechanical design requirements were valid for the operation of Cycle 14.

9.0 STARTUP TESTING

The startup testing program proposed for Cycle 14 is identical to that used in Cycle 13. It is also the same as the program outlined in the Cycle 6 Reload Application, with two exceptions. First, a CEA exchange technique (Reference 1) for zero power rod worth measurements will be performed in accordance with Reference 2, replacing the boration/dilution method. Also, low power CECOR flux maps and pseudo−ejection rod measurements will be substituted for the full core symmetry checks.

The CEA exchange technique is a method for measuring rod worths which is both faster and produces less waste than the typical boration/dilution method. The startup testing method used in Cycles 11, 12 and 13 employed the CEA exchange technique exclusively. Results from the CEA exchange technique were within the acceptance and review criteria for low power physics parameters. The combination of the pseudo−ejection technique at zero power and low power CECOR maps provides for a less time consuming but equally valid technique for detecting azimuthal power tilts during reload core physics testing. The pseudo−ejection rod measurement involves the dilution of the lead bank (Bank 4) into the core, borating a Bank 4 CEA out, and then exchanging (rod swap) the CEA against other symmetric CEA's within Bank 4 to measure rod worths.

The acceptance and review criteria for these tests are:

Test	Acceptance Criteria	Review Criteria
CEA Group Worths	± 15% of predicted	± 15% of predicted
Pseudo−ejection rod worth measurement	None	± 1.5¢ deviation from group average
Low Power CECOR maps	Technical Specification limits of F^T_R, F^T_{XY}, and T_q	Azimuthal tilt less than 20%.

OPPD has reviewed these tests and has concluded that no unreviewed safety question exists for implementation of these procedures.

10.0 REFERENCES

References (Chapters 1–5)

1. "Omaha Public Power District Reload Core Analysis Methodology Overview", OPPD–NA–8301–P, Revision 04, April 1991.
2. "Omaha Public Power District Reload Core Analysis Methodology – Neutronics Design Methods and Verification", OPPD–NA–8302–P–A, Revision 02, April 1988.
3. "Omaha Public Power District Reload Core Analysis Methodology – Transient and Accident Methods and Verification", OPPD–NA–8303–P, Revision 03, March 1991.
4. "Omaha Batch M Reload Fuel Design Report", CEN–347(O)–P Revision 1–P, January 1987.
5. "Westinghouse Reload Fuel Mechanical Design Evaluation for the Fort Calhoun Station Unit 1", WCAP–12977 (Proprietary), June 1991.

10.0 REFERENCES (Continued)

References (Chapter 6)

1. "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core", CENPD-161-P, July 1975.
2. "Critical Heat Flux Correlation For CE Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution", CENPD-152-PA April 1975.
3. "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2", CEN-191-(B)-P, December 1981
4. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 70 to Facility Operating License No. DPR-40 for the Omaha Public Power district, Fort Calhoun Station, Unit No. 1, Docket No. 50-285, March 15, 1983.
5. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 77 to Facility Operating License No. DPR-40 for the Omaha Public Power District, Fort Calhoun Station, Unit No. 1, Docket No. 50-285, April 25, 1984.
6. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 92 to Facility Operating License No. DPR-40 for the Omaha Public Power District, Fort Calhoun Station, Unit No. 1, Docket No. 50-285, November 29, 1985.
7. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 109 to Facility Operating License No. DPR-40 for Omaha Public Power District, Fort Calhoun Station, Unit No. 1, Docket No. 50-285, May 4, 1987.
8. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 117 to Facility Operating License No. DPR-40 for Omaha Public Power District, Fort Calhoun Station, Unit No. 1, Docket No. 50-285, December 14, 1988.
9. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 126 to Facility Operating License No. DPR-40 for Omaha Public Power District, Fort Calhoun Station, Unit No. 1, Docket No. 50-285, April 4, 1990.
10. "Omaha Public Power District Reload Core Analysis Methodology Overview", OPPD-NA-8301-P, Revision 04, March 1991.
11. "Statistical Combination of Uncertainties, Part 2, "Supplement 1-P, CEN-257(O)-P, August 1985.
12. Safety Evaluation Report on CENPD-207-P-A, "CE Critical Heat Flux: Part 2 Non-Uniform Axial Power Distribution", letter, Cecil Thomas (NRC) to A. E. Scherer (Combustion Engineering), November 2, 1984.

10.0 REFERENCES (Continued)

References (Chapter 7)

1. "Amendment No. 117 to Operating License DPR−40, Cycle 12 License Application", Docket No. 50−285, December 14, 1988.
2. "Statistical Combination of Uncertainties Methodology, Part 1: Axial Power Distribution and Thermal Margin/Low Pressure LSSS for Fort Calhoun", CEN−257(0)−P, November 1983."Supplement 1−P, CEN−257(O)−P, August 1985.
3. "Statistical Combination of Uncertainties Methodology, Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analysis for Fort Calhoun Unit 1", CEN−257(0)−P, November 1983.
4. "Statistical Combination of Uncertainties Methodology, Part 3: Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for Fort Calhoun", CEN−257(0)−P, November 1983.
5. "Statistical Combination of Uncertainties, Part 2, "Supplement 1−P, CEN−257(O)−P, August 1985.
6. "Omaha Public Power District Reload Core Analysis Methodology − Transient and Accident Methods and Verification", OPPD−NA−8303−P, Revision 03, March 1991.
7. "CE Setpoint Methodology", CENPD−199−P−A, Rev. 1−P, March 1982.
8. "CEA Withdrawal Methodology", CEN−121(B)−P, November 1979.
9. "CESEC, Digital Simulation of a Combustion Engineering Nuclear Steam Supply System", Enclosure 1−P to LD−82−001, January 6, 1982.
10. "Response to Questions on CESEC", Louisiana Power and Light Company, Waterford Unit 3, Docket 50−382, CEN−234(C)−P, December 1982.
11. Letter LIC−86−675, R. L. Andrews (OPPD) to A. C. Thadani (NRC), dated January 16, 1987.
12. "Omaha Public Power District Reload Core Analysis Methodology − Neutronics Design Methods and Verification", OPPD−NA−8302−P−A, Revision 02, April 1988.
13. Letter LIC−89−1172, K. J. Morris (OPPD) to Document Control Desk (NRC), dated November 8, 1989.
14. Letter LIC−91−198R, W. G. Gates (OPPD) to Document Control Desk (NRC), dated July 31, 1991.