Omaha Public Power District
Fort Calhoun Station
Unit No. 1

Cycle 16
Reload Evaluation

FORT CALHOUN STATION UNIT NO. 1
CYCLE 16
RELOAD EVALUATION

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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 sixteenth fuel cycle at a full rated power of 1,500 MWt. Planned operating conditions remain the same as those for Cycle 15, unless otherwise noted.

The core will consist of 85 presently operating Batches R and S assemblies and 48 fresh Batch T assemblies.

The Cycle 16 analysis is based on a Cycle 15 termination point between 12,800 and 14,000 MWD/MTU. In performing analyses of design basis events, limiting safety system settings and limiting conditions for operation, key parameters were chosen to assure that expected Cycle 16 conditions would be enveloped. The analysis presented herein will accommodate a Cycle 16 length of up to 14,180 MWD/MTU with a coastdown of an additional 820 MWD/MTU.

The evaluation of the reload core characteristics has been conducted with respect to the Fort Calhoun Station Unit No. 1 Cycle 15 safety analysis described in the 1994 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, changes are being incorporated into the Cycle 16 Core Operating Limits Report.

The Cycle 16 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 minimize the amount of radiation embrittlement occurring in the reactor vessel and limiting welds during this fuel cycle, based on calculations performed in accordance with Regulatory Guide 1.99, Revision 2, and 10 CFR 50.61.

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

2.0 OPERATING HISTORY OF CYCLE 15

Fort Calhoun Station is presently operating in its fifteenth fuel cycle utilizing Batches N, P, R, and S fuel assemblies. Fort Calhoun Cycle 15 operation began when criticality was achieved on November 24, 1993, and full power operation was achieved on December 3, 1993. 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 15 will be terminated on or about March 11, 1995. The Cycle 15 termination point can vary between 12,800 MWD/MTU and 14,000 MWD/MTU and still be within the assumptions of the Cycle 16 analyses. As of December 7, 1994, the Cycle 15 core average burnup had reached 11,120 MWD/MTU.

3.0 GENERAL DESCRIPTION

The Cycle 16 core will consist of the number and type of assemblies and fuel batches shown in Table 3–1. During the upcoming refueling outage, 16 Batch N assemblies, 24 Batch P assemblies and 8 Batch R assemblies will be discharged. They will be replaced by 12 fresh Batch T1 assemblies (4.15 w/o average enrichment), 12 fresh Batch T3 assemblies (4.15 w/o average enrichment with 48 IFBA rods at 0.003 gm B_{10}/in.), 16 fresh Batch T7 assemblies (3.75 w/o average enrichment with 48 IFBA rods at 0.003 gm B_{10}/in.), and 8 fresh Batch T8 assemblies (3.75 w/o average enrichment with 64 IFBA rods at 0.003 gm B_{10}/in.).

Figure 3–1 shows the fuel management loading pattern and initial enrichments to be employed in Cycle 16. The fuel management strategy for Cycle 16 is the same strategy used in Cycle 15. The overall fuel management scheme is designed to minimize the neutron leakage seen by the reactor vessel and limiting vessel weld locations. This strategy is called "extreme low radial leakage fuel management" and is the same fuel management strategy previously used in Cycles 10, 14 and 15 core loading patterns. Listed below are the key parameters that comprise the extreme low radial leakage fuel management strategy:

1) Twelve twice–burned 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 these peripheral fuel assemblies.

2) Four twice–burned natural uranium fuel assemblies will be 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) An integral fuel burnable absorber (IFBA) will continue to be used 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 (ZrB_2) coating which surrounds the fuel pellet circumferential surface area. 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 1,500 MWt. This will gain radial peaking factor margin that is needed to absorb the inward roll of the core power distribution during early cycle operation caused, in part, by the peripheral flux reduction.

3.0 GENERAL DESCRIPTION (Continued)

All fuel assemblies in the Cycle 16 core loading pattern employ multiple intra–assembly initial U–235 enrichments except for the Batch R1 naturally enriched fuel assemblies (Figure 3–2). 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 16 design exposure. Figures 3–3 through 3–7 provide diagrams of each type of Batch R, S, and T assembly that contain IFBA rods.

The average initial enrichment of the 48 fresh Batch T assemblies is 3.95 w/o U–235, an increase of 0.43 w/o from Cycle 15. For the fourth consecutive cycle, the fuel assembly zone loading technique is used in the fresh fuel assemblies to lower the radial power peaking factors. Batches T1 and T3 have fuel rods at both 4.3 w/o enriched U–235 and 3.8 w/o enriched U–235, while Batches T7 and T8 have fuel rods at both 3.9 w/o enriched U–235 and 3.4 w/o enriched U–235.

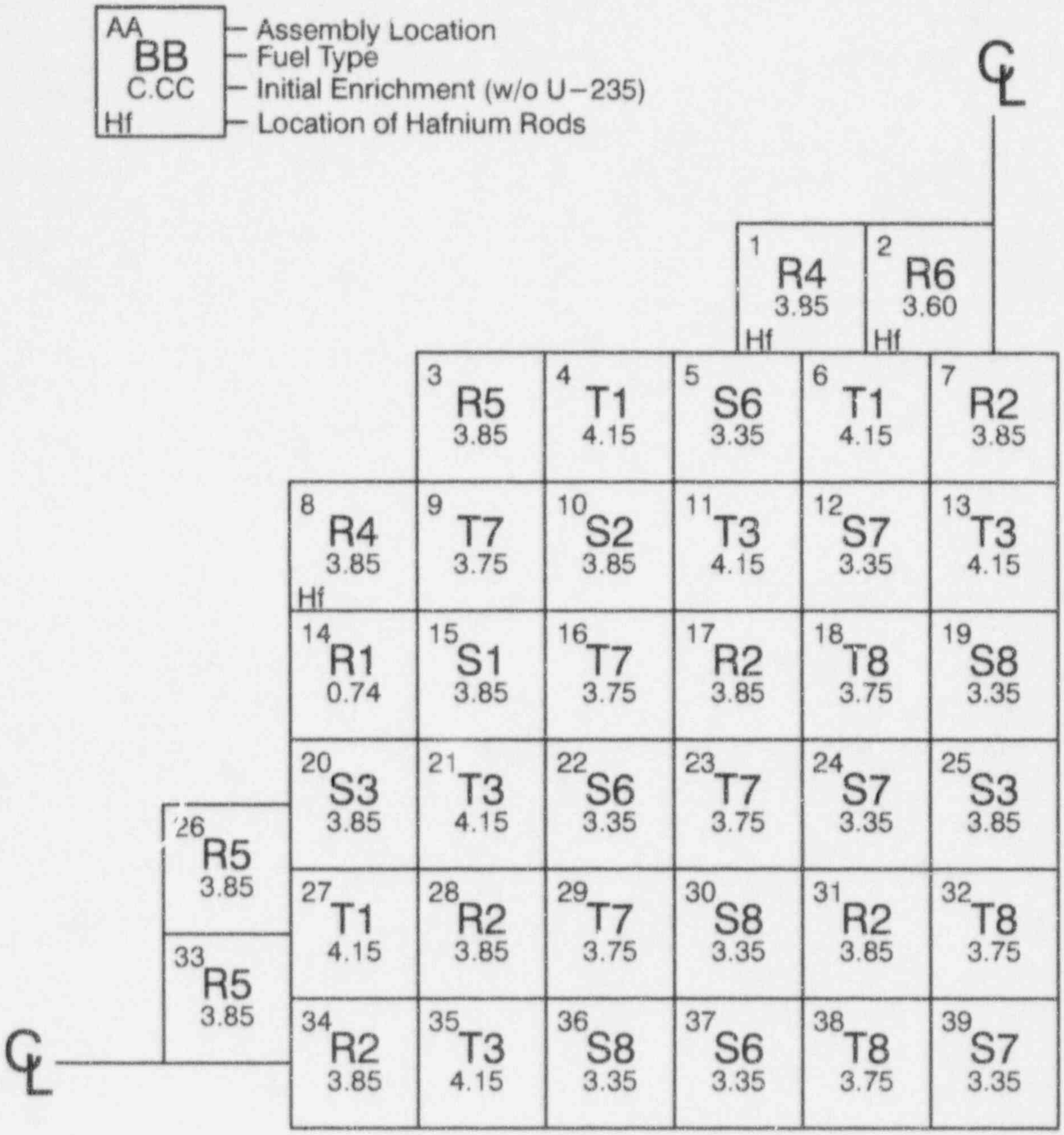
Figure 3–8 shows the beginning and end of Cycle 16 assembly burnup distributions for a Cycle 15 termination burnup of 14,000 MWD/MTU. The end of Cycle 16 core average exposure, including coastdown, will be approximately 30,449 MWD/MTU, which corresponds to a cycle average exposure of 15,000 MWD/MTU.

TABLE 3-1

FORT CALHOUN STATION UNIT NO. 1
CYCLE 16 CORE LOADING

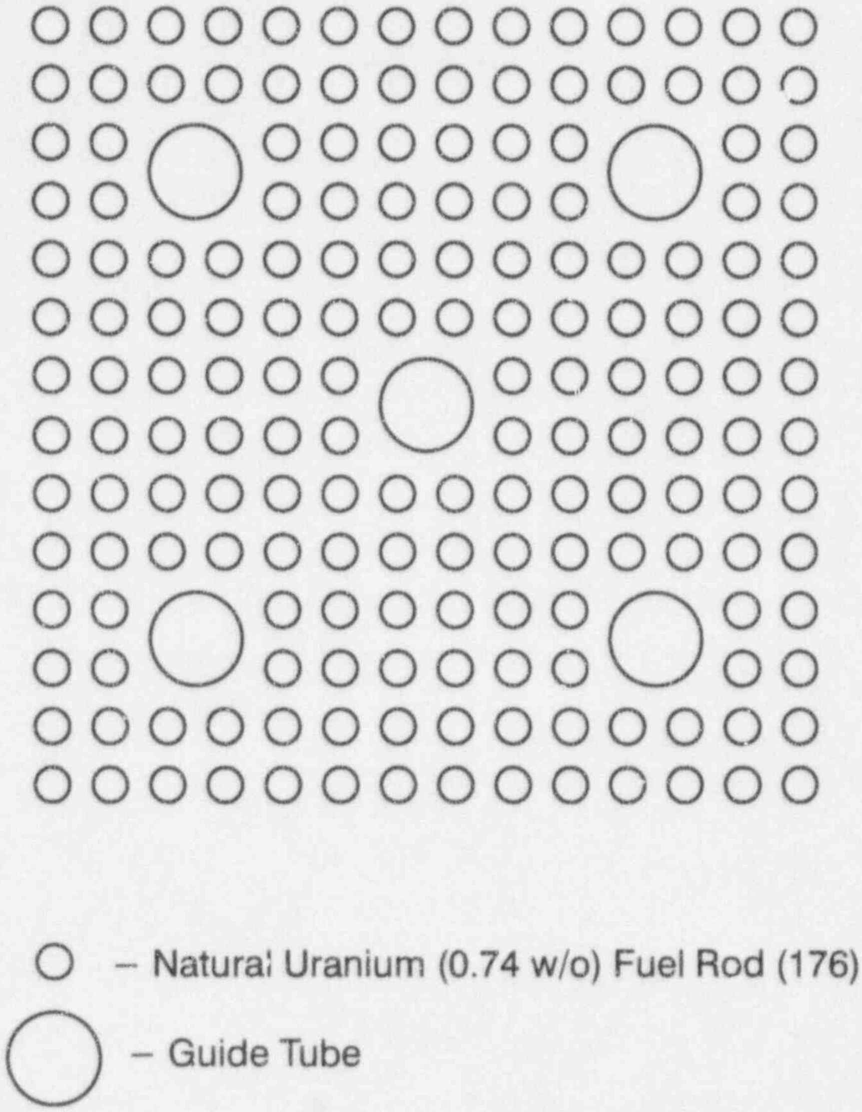
Assembly Designation	Number of Assemblies	Enrich. (Avg. w/o)	BOC Avg. Burnup[1] (MWD/MTU)	EOC Avg. Burnup[2] (MWD/MTU)	IFBA Rods per Assembly	Initial Poison Loading (gm B_{10}/in.)
R1	4	0.74	8,605	13,400	0	--
R2	16	3.85	27,585	43,534	28	0.003
R4	8	3.85	33,662	36,817	64	0.003
R5	12	3.85	34,966	40,008	84	0.003
R6	4	3.60	36,057	39,255	84	0.003
S1	4	3.85	11,609	28,310	0	–
S2	4	3.85	16,565	33,640	28	0.003
S3	6	3.85	19,914	33,665	48	0.003
S6	10	3.35	17,420	33,098	28	0.003
S7	9	3.35	19,427	36,980	48	0.003
S8	8	3.35	20,493	37,998	64	0.003
T1	12	4.15	0	15,595	0	–
T3	12	4.15	0	21,658	48	0.003
T7	16	3.75	0	20,294	48	0.003
T8	8	3.75	0	22,328	64	0.003

1 Assumes EOC15=14,000 MWD/MTU
2 Assumes EOC16=15,000 MWD/MTU

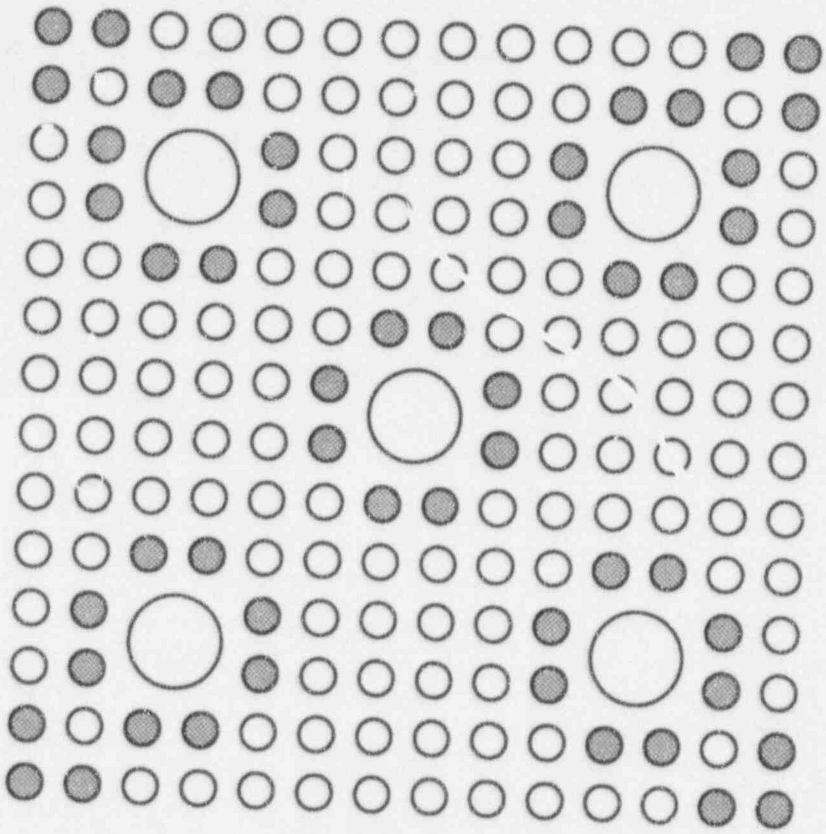


Note: EOC 15 Core Average Burnup = 14,000 MWD/MTU

Cycle 16 Core Loading Pattern and Initial Enrichments	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3−1

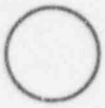


Batch R1 Assembly Fuel Rod Locations	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3–2



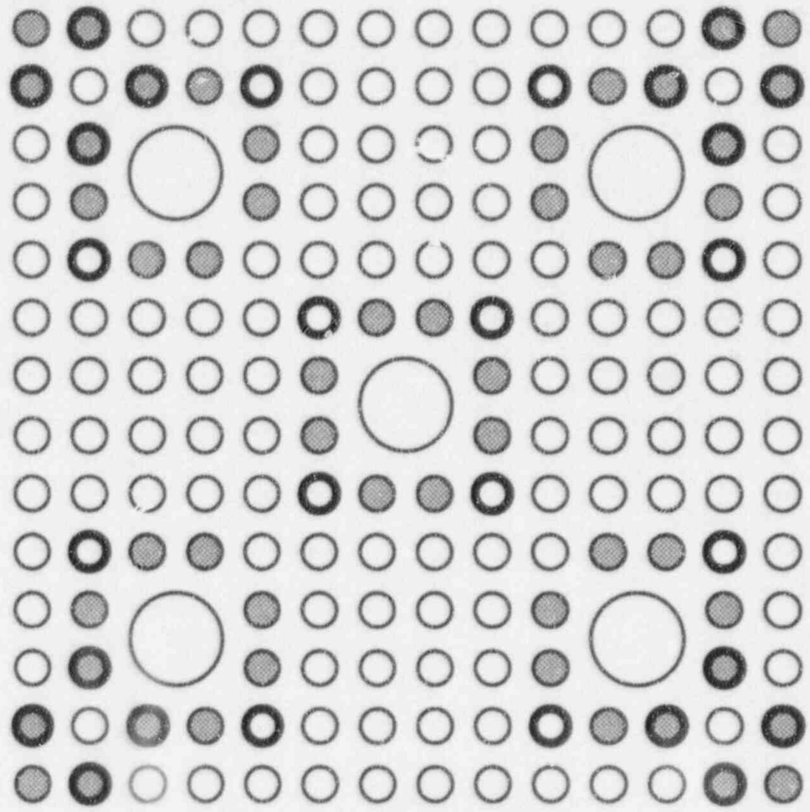
 – Low Enrichment Fuel Rod (52)

 – High Enrichment Fuel Rod (124)

 – Guide Tube

Batch	Low Enrichment (w/o)	High Enrichment (w/o)
S1	3.50	4.00
T1	3.80	4.30

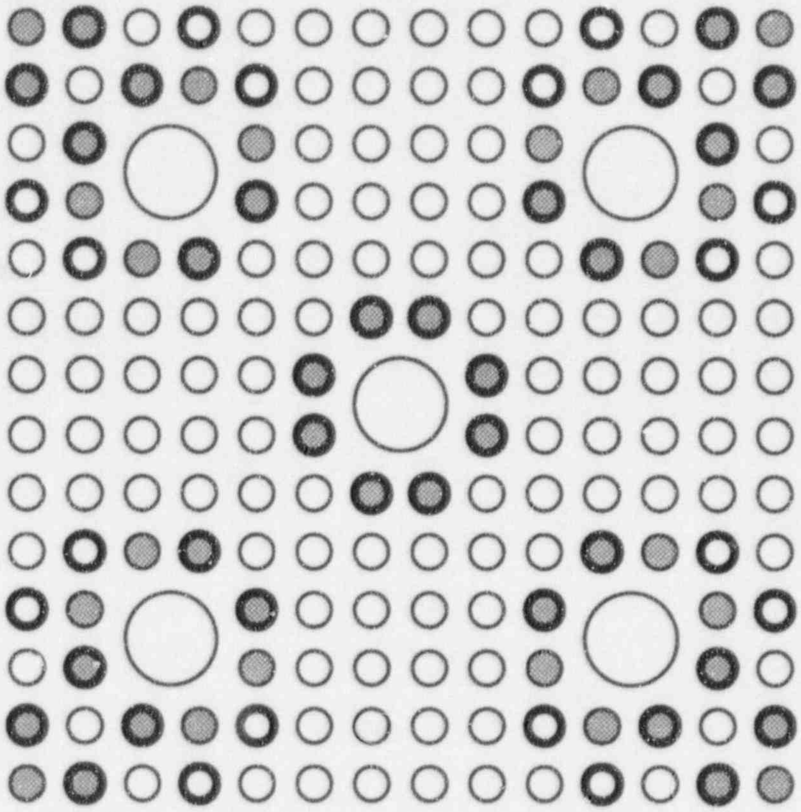
Batches S1 and T1 Assembly Fuel Rod Locations	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3–3



- Low Enrichment Fuel Rod (36)
- Low Enrichment Fuel Rod with IFBA (16)
- High Enrichment Fuel Rod (112)
- High Enrichment Fuel Rod with IFBA (12)
- Guide Tube

Batch	Low Enrichment (w/o)	High Enrichment (w/o)
R2	3.50	4.00
S2	3.50	4.00
S6	3.00	3.50

Batches R2, S2 and S6 Assembly Fuel Rod and 28 IFBA Rod Locations	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3-4



– Low Enrichment Fuel Rod (20)

– Low Enrichment Fuel Rod with IFBA (32)

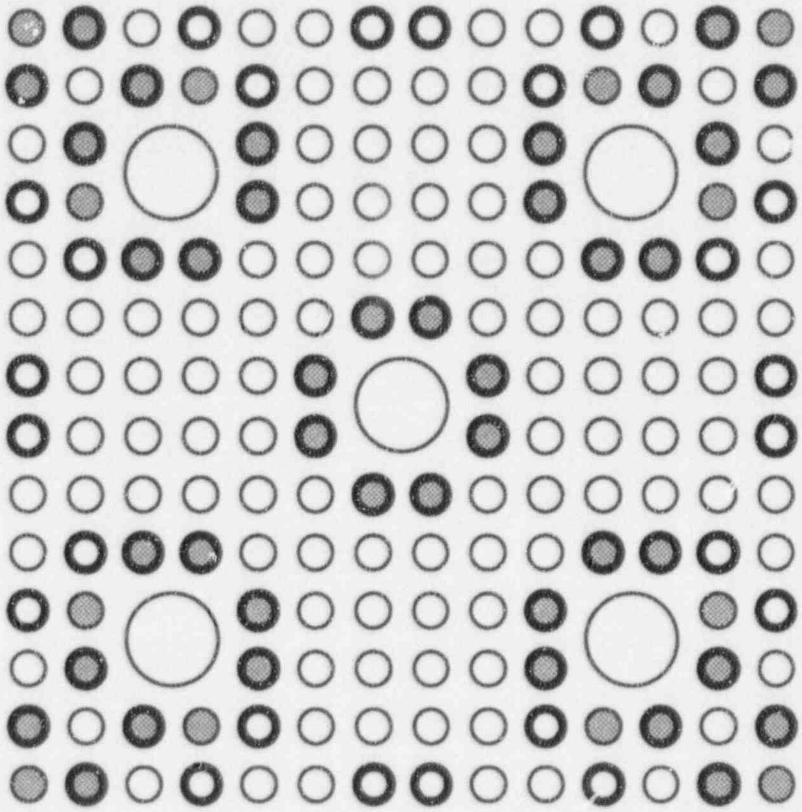
– High Enrichment Fuel Rod (108)

– High Enrichment Fuel Rod with IFBA (16)

– Guide Tube

Batch	Low Enrichment (w/o)	High Enrichment (w/o)
S3	3.50	4.00
S7	3.00	3.50
T3	3.80	4.30
T7	3.40	3.90

Batches S3, S7, T3 and T7 Assembly Fuel Rod and 48 IFBA Rod Locations	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3–5



– Low Enrichment Fuel Rod (12)

– Low Enrichment Fuel Rod with IFBA (40)

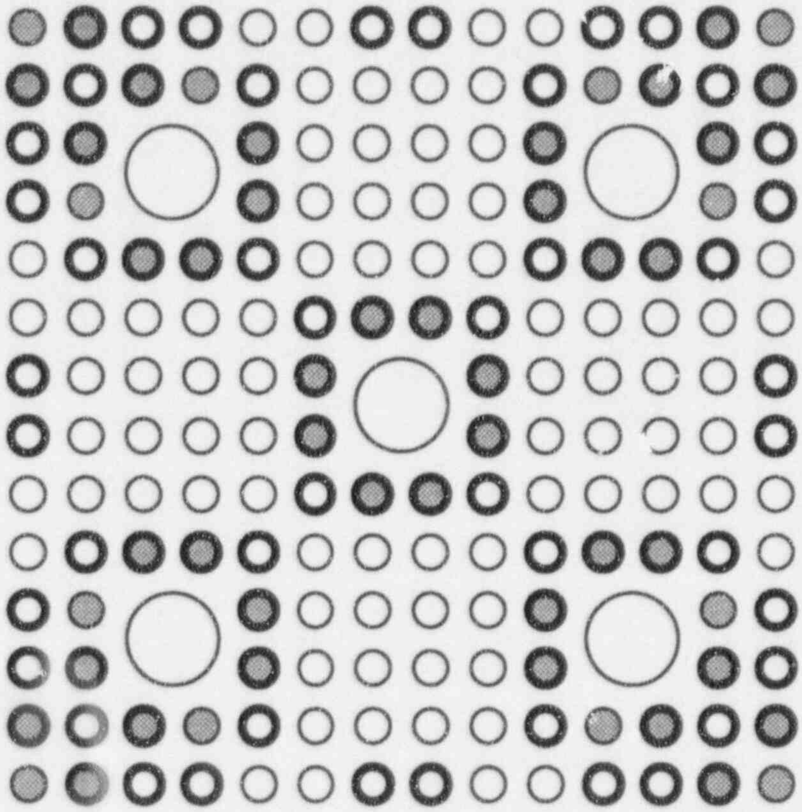
– High Enrichment Fuel Rod (100)

– High Enrichment Fuel Rod with IFBA (24)

– Guide Tube

Batch	Low Enrichment (w/o)	High Enrichment (w/o)
R4	3.50	4.00
S8	3.00	3.50
T8	3.40	3.90

Batches R4, S8 and T8 Assembly Fuel Rod and 64 IFBA Rod Locations	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3–6



– 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	3.50	4.00
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 3–7

Key	
AA	Assembly Location
BB	Fuel Type
CC,CCC	BOC Assembly Average Exposure (MWD/MTU)
DD,DDD	EOC Assembly Average Exposure (MWD/MTU)

Assembly Location	Fuel Type	BOC Exposure (MWD/MTU)	EOC Exposure (MWD/MTU)
1	R4	29,550	32,870
2	R6	36,015	39,214
3	R5	37,566	42,612
4	T1	0	13,270
5	S6	13,763	26,187
6	T1	0	16,459
7	R2	27,734	39,484
8	R4	37,776	40,768
9	T7	0	13,758
10	S2	16,575	33,648
11	T3	0	21,884
12	S7	20,605	37,493
13	T3	0	21,995
14	R1	8,606	13,401
15	S1	11,608	28,310
16	T7	0	22,128
17	R2	29,323	46,928
18	T8	0	22,480
19	S8	20,313	38,122
20	S3	19,735	31,134
21	T3	0	21,173
22	S6	19,647	37,974
23	T7	0	22,849
24	S7	18,403	36,369
25	S3	20,272	38,732
26	R5	29,568	34,368
27	T1	0	17,070
28	R2	26,700	43,385
29	T7	0	22,438
30	S8	20,629	37,983
31	R2	25,445	42,951
32	T8	0	22,287
33	R5	37,782	43,063
34	R2	30,025	42,275
35	T3	0	21,840
36	S8	20,397	37,903
37	S6	20,294	37,183
38	T8	0	22,065
39	S7	18,813	37,379

Note: EOC 15 Core Average Burnup = 14,000 MWD/MTU
EOC 16 Core Average Burnup = 15,000 MWD/MTU

Cycle 16 BOC and EOC Assembly Average Exposure	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3–8

4.0 FUEL SYSTEMS DESIGN

The mechanical design for the Batch T fuel is the same as the Batches R and S fuel supplied by Westinghouse in Cycles 14 and 15, respectively.

The Batch T fuel is mechanically, thermally, and hydraulically compatible with the Westinghouse fuel returning to the Cycle 16 core. The Batch T fuel incorporates an extra level of defense against debris–induced damage with a hardened coating of zirconium dioxide (ZrO_2) surrounding the bottom 6 inches of each fuel rod. Should debris pass through the bottom nozzle and progress through the lower dimples of the bottom grid, this ZrO_2 coating provides an added measure of protection. This ZrO_2 layer is twice as hard as most common types of debris and will increase wear resistance by a factor of 10 over the cladding used in Batches R and S fuel.

Reference 4 describes the Westinghouse fuel characteristics and design. During Cycle 16, the Westinghouse fuel will not be resident in the reactor with any of the Exxon (i.e., Siemens) or ABB–Combustion Engineering fuel previously used at Fort Calhoun Station.

5.0 NUCLEAR DESIGN

5.1 PHYSICAL CHARACTERISTICS

5.1.1 Fuel Management

Cycle 16 fuel management uses an extreme low radial leakage design, with twice burned assemblies predominantly loaded on the periphery of the core and 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 fast neutron flux to the pressure vessel welds and achieve the maximum in neutron economy. Use of this type of fuel management to achieve significant reduction in pressure vessel neutron flux over a standard out–in–in pattern results in higher radial peaking factors. The maximum radial peaking factors for Cycle 16 have been minimized 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 16 loading pattern which is composed of 48 fresh Batch T assemblies of which 36 contain the aforementioned IFBA pellet design. All of these 48 assemblies employ intra–assembly uranium enrichment splits. Batches T1 and T3 contain a high pin U–235 enrichment of 4.30 w/o and a low pin U–235 enrichment of 3.80 w/o, while Batches T7 and T8 contain a high pin U–235 enrichment of 3.90 w/o and a low pin U–235 enrichment of 3.40 w/o. Forty twice burned R assemblies are being returned to the core along with 41 once burned S assemblies. Four of the returning Batch R assemblies contain fuel rods that are loaded with naturally enriched uranium and placed in locations near the limiting reactor vessel welds. This assembly arrangement will produce a Cycle 16 loading pattern with a cycle energy of 14,180 MWD/MTU with an additional 820 MWD/MTU of energy in a coastdown mode if required. The Cycle 16 core characteristics have been examined for a Cycle 15 termination ranging from 12,800 to 14,000 MWD/MTU with limiting values established for the safety analysis.

Nominal physics parameters, including reactivity coefficients for Cycle 16, are listed in Table 5–1 along with the corresponding values from Cycle 15. 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 were the most limiting conditions used in the determination of available shutdown margin for compliance with the Technical Specifications. For Cycle 16, the minimum available scram worth/shutdown margin is 1.08 %$\Delta\rho$ greater than the Technical Specification requirement of 4.0 %$\Delta\rho$.

5.0 NUCLEAR DESIGN (Continued)

5.1 PHYSICAL CHARACTERISTICS (Continued)

5.1.1 Fuel Management (Continued)

Table 5–2 presents a summary of HZP CEA shutdown worths and reactivity allowances for Cycle 16. The Cycle 16 CEA worth values, used in the calculation of minimum scram worth, exceed the minimum value of 4.0 $\%\Delta\rho$ currently required by Technical Specifications and thus provide adequate shutdown margin.

5.1.2 Power Distribution

Figure 5–1 illustrates the all rods out (ARO) planar radial power distributions at BOC16, MOC16, and EOC16 and is based upon the Cycle 15 late window burnup of 14,000 MWD/MTU. These relative power densities are assembly averages representative of the entire core length. The high burnup end of the Cycle 15 shutdown window tends to increase the power peaking in the high power assemblies in the Cycle 16 fuel loading pattern. The radial power distributions, with Bank 4 fully inserted at beginning and end of Cycle 16, are shown in Figure 5–2.

The radial power distributions described in this section are derived from calculated data without uncertainties or other allowances with the exception of the single rod power peaking values. The power peaking values used for the DNB and kW/ft safety and setpoint analyses conservatively bound all Cycle 16 predictions for both unrodded and rodded configurations. 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, Figure 3–8 shows the integrated assembly burnup values at 0 and 15,000 MWD/MTU for Cycle 16.

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 16 during normal base load, ARO operation at full power is 2.11, 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 16. These bounding values are projected to encompass the worst conditions anticipated during Cycles 14 through 18 operation and were calculated in accordance with Reference 3.

5.1.3.2 Dropped CEA Data

The Cycle 16 safety related data for the dropped CEA analysis were calculated identically with the methods used in Cycle 15.

5.2 ANALYTICAL INPUT TO INCORE MEASUREMENTS

Incore detector measurement constants to be used in the generation of the Cycle 16 power distributions will be calculated using a method similar to Cycle 15. These constants will be based upon power distribution information generated by the SIMULATE−3 code.

5.3 NUCLEAR DESIGN METHODOLOGY

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

5.4 UNCERTAINTIES IN MEASURED POWER DISTRIBUTIONS

The power distribution measurement uncertainties applied to Cycle 16 are the same as those presented in Reference 2.

TABLE 5-1

FORT CALHOUN STATION UNIT NO. 1, CYCLE 16
NOMINAL ALL RODS OUT PHYSICS CHARACTERISTICS

	Units	Cycle 15	Cycle 16
Critical Boron Concentration			
HFP, BOC, Equilibrium Xenon	ppm	978	1055
Inverse Boron Worth			
HFP, BOC	ppm/%$\Delta\rho$	117	124
HFP, EOC	ppm/%$\Delta\rho$	91	96
Moderator Temperature Coefficient (MTC)*			
HZP, BOC	x 10^{-4} $\Delta\rho$/°F	+0.34	+0.40
HFP, EOC	x 10^{-4} $\Delta\rho$/°F	−2.84	−2.93
Doppler Coefficient (FTC)			
HFP, BOC	x 10^{-5} $\Delta\rho$/°F	−1.60	−1.59
HFP, EOC	x 10^{-5} $\Delta\rho$/°F	−1.77	−1.63
Total Delayed Neutron Fraction (β_{eff})			
HFP, BOC		0.00620	0.00625
HFP, EOC		0.00512	0.00520
Neutron Generation Time (l*)			
HFP, BOC	10^{-6} sec	20.8	19.7
HFP, EOC	10^{-6} sec	26.8	24.8

* Includes uncertainties.

TABLE 5-2

FORT CALHOUN STATION UNIT NO. 1, CYCLE 16
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	7.78	8.69
2. Stuck CEA Allowance	1.31	1.43
3. Worth of all CEAs Less Worth of Most Reactive CEA Stuck Out	6.47	7.26
4. Power Dependent Insertion Limit CEA Worth	1.04	1.18
5. Calculated Scram Worth	5.43	6.08
6. Physics Uncertainty Plus Bias	0.35	0.40
7. Net Available Scram Worth	5.08	5.68
8. Technical Specification Shutdown Margin	4.00	4.00
9. Margin in Excess of Technical Specification Shutdown Margin	1.08	1.68

TABLE 5–3

FORT CALHOUN STATION UNIT NO. 1
BOUNDING CEA EJECTION DATA

Maximum Radial Power Peaking Factor	Bounding Value
Full Power with Bank 4 Inserted and Worst CEA Ejected	3.73
Zero Power with Banks 4+3 Inserted and Worst CEA Ejected	5.74

Maximum Ejected CEA Worth (%$\Delta\rho$)	Bounding Value
Full Power with Bank 4 Inserted and Worst CEA Ejected	0.36
Zero Power with Banks 4+3 Inserted and Worst CEA Ejected	0.69

NOTE: The above bounding values encompass all conditions between BOC and EOC and include applicable biases and uncertainties.

AA	Assembly Location					
B.BBBB	Assembly Relative Power Density at BOC					
C.CCCC	Assembly Relative Power Density at 7,000 MWD/MTU					
D.DDDD	Assembly Relative Power Density at 15,000 MWD/MTU					

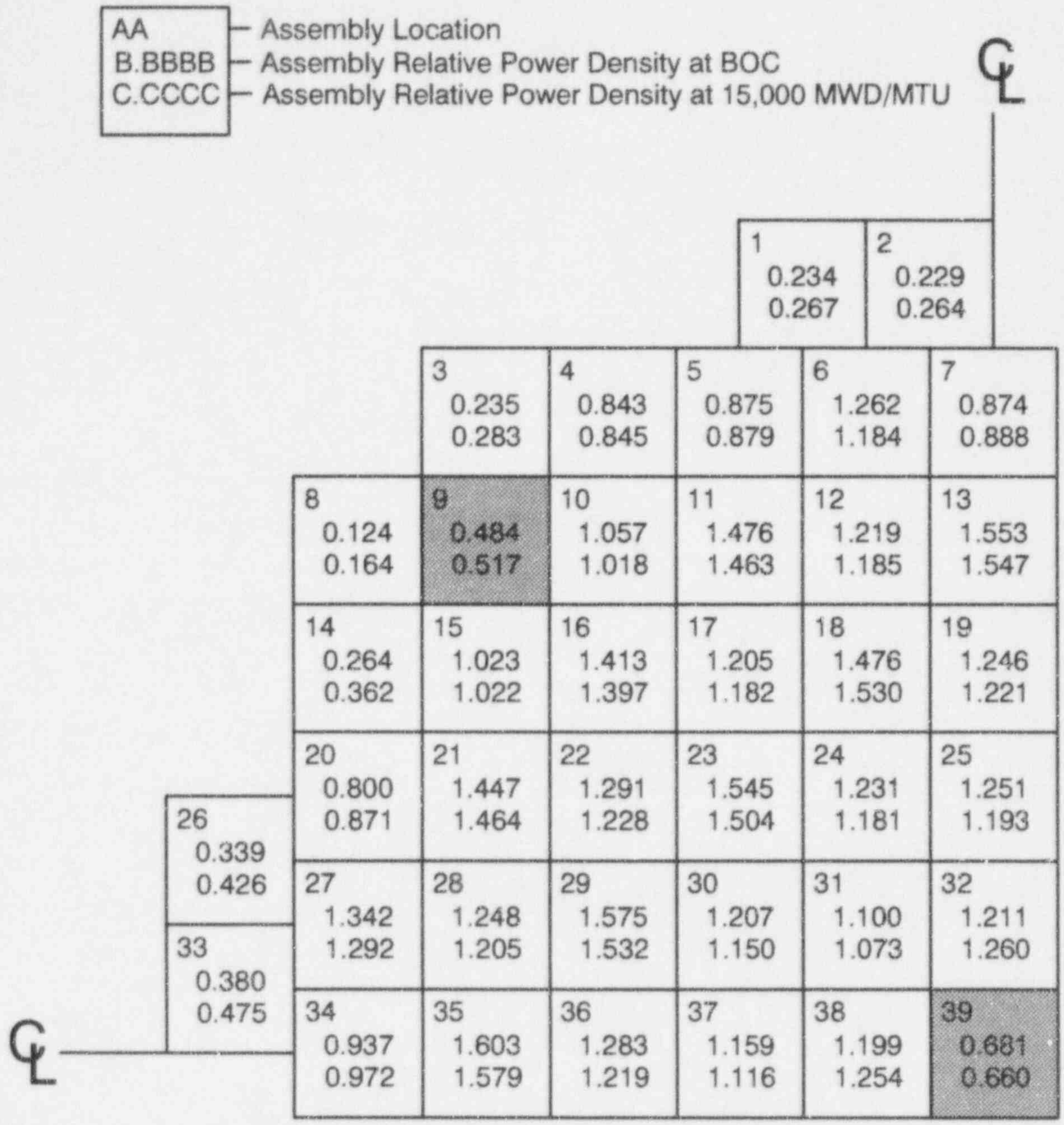
CL

				1 0.222 0.215 0.249	2 0.214 0.207 0.241	
		3 0.305 0.326 0.400	4 0.905 0.868 0.937	5 0.857 0.816 0.859	6 1.190 1.080 1.098	7 0.815 0.771 0.811
	8 0.172 0.193 0.245	9 0.833 0.909 1.015	10 1.151 1.132 1.145	11 1.448 1.470 1.430	12 1.154 1.128 1.104	13 1.458 1.480 1.423
	14 0.282 0.309 0.397	15 1.108 1.104 1.142	16 1.439 1.493 1.435	17 1.172 1.183 1.138	18 1.411 1.533 1.436	19 1.187 1.197 1.138
26 0.313 0.304 0.386	20 0.764 0.740 0.828	21 1.401 1.413 1.412	22 1.250 1.227 1.178	23 1.497 1.549 1.435	24 1.209 1.210 1.139	25 1.244 1.241 1.165
33 0.347 0.334 0.424	27 1.245 1.107 1.177	28 1.169 1.101 1.109	29 1.496 1.510 1.430	30 1.181 1.160 1.107	31 1.158 1.180 1.117	32 1.366 1.528 1.413
	34 0.861 0.792 0.875	35 1.490 1.453 1.438	36 1.212 1.164 1.129	37 1.147 1.129 1.088	38 1.353 1.511 1.407	39 1.178 1.267 1.176

CL

Maximum 1-Pin Peak	Q.C. Assembly	% of Core Height
1.727	35	16
1.773	13	22
1.675	21	16

Cycle 16 Assembly Power Distribution ARO, HFP, Eq. Xenon	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 5-1



Maximum 1-Pin Peak	Q.C. Assembly	% of Core Height
1.868	35	22
1.832	35	16

▨ – Bank 4 Location

Cycle 16 Assembly Power Distribution Bank 4 In, HFP, Eq. Xenon	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 5-2

6.0 THERMAL–HYDRAULIC DESIGN

6.1 DNBR ANALYSIS

Steady state DNBR analyses for Cycle 16, 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 as was done for the Cycles 14 and 15 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 analysis methodology for Cycle 16 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 minimum DNBR of random variations in spacing between fuel rods. The penalty at 45,000 MWD/MTU burnup is 0.5% in minimum DNBR. This penalty was applied in the derivation of the SCU minimum DNBR design limit of 1.18 (References 6 and 12) in the statistical combination of uncertainties (Reference 11). 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. 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 STATION UNIT NO. 1, CYCLE 16
THERMAL HYDRAULIC PARAMETERS AT FULL POWER

	Unit	Cycle 16*
Total Heat Output (Core Only)	MWt	1,500
	10^6 BTU/hr	5,119.5
Fraction of Heat Generated in Fuel Rod		0.975
Primary System Pressure		
Nominal	psia	2,100
Minimum In Steady State	psia	2,075
Maximum In Steady State	psia	2,150
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	0.044
Average Mass Velocity	10^6 lbm/hr-ft^2	2.2254
Core Average Heat Flux		
(Accounts for Heat Generated in Fuel Rod)	BTU/hr-ft^2	177,997
Total Heat Transfer Surface Area	ft^2	28,761.7
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.

** 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 16 Non-LOCA safety analyses at 1500 MWt.

The Design Bases Events (DBEs) considered in the safety analysis are listed in Table 7.0-1. 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), is 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 7.0-2.

As indicated in Table 7.0-1, no reanalysis was performed for the DBEs for which key transient input parameters are within the bounds of (i.e., conservative with respect to) the reference cycle values (Fort Calhoun Station, Unit 1, Updated Safety Analysis Report including Cycle 15 analyses, Reference 1). For these DBEs the results and conclusions quoted in the reference cycle analysis remain valid for Cycle 16.

For those analyses indicated as reviewed, calculations were performed in accordance with Reference 6 until a determination could be made that Cycle 16 results would be bounded by Cycle 15 or the USAR reference cycle.

Events were evaluated for up to a total of 6% steam generator tube plugging. Use of the 6% tube plugging has been shown to be conservative since Cycle 11. Fort Calhoun Station currently has 1.09% steam generator tubes plugged; thus, no additional analysis is required.

For the events reanalyzed, Table 7.0-3 shows the reason for the reanalysis, the acceptance criteria 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 7.0-1

FORT CALHOUN STATION UNIT NO. 1, CYCLE 16
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
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]
7.1.6	Sequential CEA Group Withdrawal (LHR)	Reanalyzed

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 (DNB)	Reanalyzed
7.2.3	Loss of Coolant Flow	Reanalyzed[3]
7.2.4	CEA Drop	Reanalyzed
7.2.5	Boron Dilution	Reanalyzed
7.2.6	Transients Resulting from the Malfunction of One Steam Generator	Not Reanalyzed[4]

7.3 Postulated Accidents

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

[1] Technical Specifications preclude this event during operation.

[2] Requires High Power/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.

TABLE 7.0-2

FORT CALHOUN STATION UNIT NO. 1, CYCLE 16
CORE PARAMETERS INPUT TO SAFETY ANALYSES
FOR DNB AND CTM (CENTERLINE TO MELT) DESIGN LIMITS

Parameter	Units	Cycle 15	Cycle 16
F_R^T for DNB Margin Analyses			
Unrodded Region		1.767*	1.770*
Bank 4 Inserted		1.886*	1.887*
F_{XY}^T for Planar Radial Component of 3-D Peak (CTM Limit Analyses)			
Unrodded Region		1.854*	1.859*
Bank 4 Inserted		1.975*	1.994*
Maximum Augmentation Factor	$10^{-4}\ \Delta\rho/°F$	1.000	1.000
Moderator Temperature Coefficient	$10^{-4}\ \Delta\rho/°F$	−3.0 to +0.5	−3.0 to +0.5
Shutdown Margin (Value Assumed in Limiting EOC Zero Power SLB)	$\%\Delta\rho$	−4.0	−4.0
Power Level	MWt	1,500**	1,500**
Maximum Steady State Temperature	°F	543**	545**
Minimum Steady State Pressurizer Pressure	psia	2,075**	2,075**
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	15.5	15.5
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 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 through 5.

** 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 through 5.

TABLE 7.0-3

FORT CALHOUN STATION UNIT NO. 1
DESIGN BASIS EVENTS REANALYZED FOR CYCLE 16

Event	Reason for Reanalysis	Acceptance Criteria	Summary of Results
Sequential CEA Group Withdrawal	To calculate cycle-specific ROPM values	Minimum DNBR $\geq$ 1.18 using the CE-1 correlation. Transient PLHGR $\leq$ 22 kW/ft or Center Line Melt Temp. < 4800 °F	MDNBR = 1.436 CTM < 4800 °F ROPM = 116.3% (at positive ASI limit)*
CEA Drop	To calculate cycle-specific ROPM values	Minimum DNBR $\geq$ 1.18 using the CE-1 correlation. Transient PLHGR $\leq$ 22 kW/ft.	MDNBR = 1.401 PLHGR < 22 kW/ft ROPM = 116.1% (at positive ASI limit)*
Excess Load	To calculate cycle-specific ROPM values	Minimum DNBR $\geq$ 1.18 using the CE-1 correlation. Transient PLHGR $\leq$ 22 kW/ft.	MDNBR = 1.383 PLHGR < 22 kW/ft ROPM = 116.3% (at negative ASI limit)*
RCS Depressurization	To provide a conservative P_{bias} input for the TM/LP	P_{bias} value $\leq$ the previous cycle's limiting value	P_{bias} = 30 psia
Loss of Coolant Flow	Non-conservative rod worths compared to the reference cycle. Calculated cycle-specific ROPM values	Minimum DNBR $\geq$ 1.18 using the CE-1 correlation. Transient PLHGR $\leq$ 22 kW/ft.	MDNBR = 1.494 PLHGR < 22 kW/ft ROPM = 110.4% (at positive ASI limit)*
Main Steam Line Break	To verify if the cooldown curve with MTC = -3.0×10^{-4} $\Delta\rho$/°F was bounded by Cycle 8 analysis and update the USAR analysis.	Return to power during the event for Cycle 16 must be bounded by the return to power calculation performed for Cycle 8 and bounded by Cycle 1.	Return to power for Cycle 16 was 11.86% vs. 12.92% for Cycle 8 which was bounded by Cycle 1.
Boron Dilution	To verify sufficient time is available for operator identification and termination of the event.	t > 15 minutes (Modes 1-4) t > 30 minutes (Mode 5)	Mode 4 (drained) was most limiting with t = 15.27 min. Mode 5: t = 42.04 min.

* Note: ROPM values are dependent upon ASI conditions. Most limiting ROPM reported including PDIL and ARO conditions.

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 16 to determine the pressure bias term for the TM/LP trip setpoint.

The RCS Depressurization event is the Design Basis Event analyzed to determine the maximum pressure bias term input to the TM/LP trip. The methodology used for Cycle 16 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. Key parameters of this analysis are shown in Table 7.1.1−1. Table 7.1.1−2 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 was reclassified in Cycle 14 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 STATION UNIT NO. 1, CYCLE 16
KEY PARAMETERS ASSUMED IN THE RCS DEPRESSURIZATION ANALYSIS

Parameter	Units	Cycle 16
Initial Core Power Level	MWt	1,530
Core Inlet Coolant Temperature	°F	547
Pressurizer Pressure	psia	2,172
Moderator Temperature Coefficient	$\times 10^{-4}\ \Delta\rho/°F$	−3.0
Fuel Temperature Coefficient	$\times 10^{-4}\ \Delta\rho/°F$	Most negative prediction during core life
Core Average Hgap	$BTU/hr-ft^2-°F$	500
Total Trip Delay Time	sec	1.4

TABLE 7.1.1-2

FORT CALHOUN STATION UNIT NO.1, CYCLE 16
SEQUENCE OF EVENTS FOR RCS DEPRESSURIZATION

Time (sec)	Event	Setpoint or Value
0.000	Inadvertent Opening of Both Pressurizer Power Operated Relief Valves	------
7.210	Reactor Trip	2,079.07 psia
8.810	Time of Minimum DNBR	2,052.26 psia

7.0 TRANSIENT ANALYSIS (Continued)

7.1 ANTICIPATED OPERATIONAL OCCURRENCES (CATEGORY 1) (Continued)

7.1.2 CEA Withdrawal Event (LHR)

A CEA Withdrawal (CEAW) event is assumed to occur as a result of a failure in the control element drive circuits or by operator error. The methodology contained in Reference 6 was employed in analyzing the CEAW event. This event is classified as one for which the acceptable fuel centerline–to–melt (CTM) limit is not violated by virtue of the Variable High Power trip (VHPT).

The CEAW event was reanalyzed for Cycle 16 to verify that the VHPT would provide sufficient margin to ensure that the CTM design limit will not be exceeded.

The centerline melt SAFDL is not exceeded if the peak linear heat generation rate (PLHGR) does not exceed its established steady state limit of 22 kW/ft. For some CEAW cases, a rapid core power rise is obtained for a short period of time. The PLHGR for these cases may exceed the steady state limit. For these cases the total energy generated and the temperature rise at the hot spot are computed for the duration of the transient to demonstrate that the fuel centerline temperature does not exceed the UO_2 melt temperature.

Hot Full and Other Power Conditions.

This event is analyzed at HFP, 62.6% power, 30.0% and HZP since the rate of reactivity insertion for CEAW events initiated from power levels less than HFP is larger than if it was initiated at HFP due to the greater insertion limit of the CEAs allowed by the COLR PDIL LCO.

The analysis shows that the fuel to centerline melt temperatures are below those corresponding to the acceptable fuel to centerline melt limit. The key input parameters used for the zero and hot full power cases are presented in Table 7.1.2–1 and 7.1.2–2. Table 7.1.2–3 indicates the sequence of events for the hot zero power case.

The event is protected by the Variable High Power Trip (VHPT), terminating further degradation in LHR margins. Additional trip protection is provided in High Power Trip (HPT), Axial Power Distribution (APD) trip, and High Pressurizer Pressure trip.

In conclusion, the CEAW event, in conjunction with the VHPT limit from the Technical Specifications, will not lead to a fuel temperature which violates the CTM design limit.

TABLE 7.1.2-1

FORT CALHOUN STATION UNIT NO. 1, CYCLE 16
KEY PARAMETERS ASSUMED IN THE HFP CEA WITHDRAWAL ANALYSIS

Parameter	Units	Cycle 16
Initial Core Power Level	MWt	1,500*
Core Inlet Coolant Temperature	°F	545*
Pressurizer Pressure	psia	2,075*
Moderator Temperature Coefficient	$\times 10^{-4}\ \Delta\rho/°F$	+0.5
Doppler Coefficient Multiplier		1.00
CEA Worth at Trip	$\%\Delta\rho$	6.061
Reactivity Insertion Rate Range	$\times 10^{-4}\ \Delta\rho/sec$	0 to 2.3
CEA Group Withdrawal Rate	in/min	46
Holding Coil Delay Time	sec	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.

TABLE 7.1.2-2

FORT CALHOUN STATION UNIT NO. 1, CYCLE 16
KEY PARAMETERS ASSUMED IN THE HZP CEA WITHDRAWAL ANALYSIS

Parameter	Units	Cycle 12	Cycle 16
Initial Core Power Level	MWt	1	1%*
Core Inlet Coolant Temperature	°F	532	532*
Pressurizer Pressure	psia	2,053	2,075*
Moderator Temperature Coefficient	$\times 10^{-4}\ \Delta\rho/°F$	+0.5	+0.5
Doppler Coefficient Multiplier		0.85	1.00
CEA Worth at Trip	$\%\Delta\rho$	5.28	6.789
Reactivity Insertion Rate Range	$\times 10^{-4}\ \Delta\rho/sec$	0 to 1.0	0 to 2.9
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.

TABLE 7.1.2-3

FORT CALHOUN STATION UNIT NO.1, CYCLE 16
SEQUENCE OF EVENTS FOR THE HZP CEA WITHDRAWAL ANALYSIS

Time (sec)	Event	Setpoint or Value
0.000	Inadvertent Withdrawal of CEAs	------
19.4	High Power Trip	30%
20.4	Peak Power Reached	64.2%
21.3	Time of Minimum DNBR	5.547

7.0 TRANSIENT ANALYSIS (Continued)

7.2 ANTICIPATED OPERATIONAL OCCURRENCES (CATEGORY 2)

7.2.1 Excess Load Event

The Excess Load event was reanalyzed for Cycle 16 to determine the DNB and LHR ROPMs which are used to ensure that sufficient margin is included in the DNB and LHR LCOs in order to protect 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 16 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 CEA Drop and CEA Withdrawal, in order 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.

The Cycle 16 analysis concludes that the ROPM required by the Excess Load event was bounded by the requirements of the CEA Withdrawal Event for ASIs that are positive. For negative ASI conditions, the Excess Load event is most limiting.

TABLE 7.2.1-1

FORT CALHOUN STATION UNIT NO. 1, CYCLE 16
KEY PARAMETERS ASSUMED IN THE EXCESS LOAD ANALYSIS

Parameter	Units	Cycle 16
Initial Core Power Level	MWt	1,500*
Core Inlet Coolant Temperature	°F	545*
Pressurizer Pressure	psia	2,075*
Moderator Temperature Coefficient	$\times 10^{-4}\ \Delta\rho/°F$	-1.4708
Doppler Coefficient Multiplier		1.00
CEA Worth at Trip	$\%\Delta\rho$	6.061
Excore Temperature Shadowing Factor	%/°F	0.5
Cold Leg RTD Time Constant	sec	12.0 (max.)
Hot Leg RTD Time Constant	sec	3.0 (min.)

* 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 7.2.1-2

FORT CALHOUN STATION UNIT NO.1, CYCLE 16
SEQUENCE OF EVENTS FOR THE EXCESS LOAD ANALYSIS

Time (sec)	Event	Setpoint or Value
0.000	Steam Dump and Bypass Valves Open	------
37.60	High Power Trip Conditions Reached	112%
38.10	High Power Trip Signal Generated	112%
38.30	Time of Minimum DNBR	1.383

7.2.2 CEA Withdrawal Event (DNB)

A CEA Withdrawal (CEAW) event is assumed to occur as a result of a failure in the control element drive circuits or by operator error. The methodology contained in Reference 6 was employed in analyzing the CEAW event. This event is classified as one for which the acceptable DNBR limit is not violated by virtue of maintenance of sufficient initial steady state thermal margin provided by the DNBR related Limiting Condition for Operation (LCO). CEA Withdrawal, with respect to DNBR, is classified as a Category Type 2 event where steady state thermal margin is incorporated into the LCO.

The CEAW event was reanalyzed for Cycle 16 to determine the initial margins that must be maintained by the LCO such that the DNBR design limit will not be exceeded.

The CEA Withdrawal Event–DNBR margin is maintained by the LCOs since sufficient steady state thermal margin is provided to prevent exceeding the acceptable limits. The DNBR margin was analyzed at HFP, 62.6% power, 30.0% power and HZP for the same reasons as specified for the LHR calculations. The key input parameters used for the zero and hot full power cases are presented in Table 7.2.2–1 and 7.2.2–2.

Hot Full and Other Power Conditions.

The HFP and other power cases for Cycle 16 are 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 LCOs.

The zero power case initiated at the limiting conditions for operation results in a minimum CE–1 DNBR of 5.547 which is less than the Cycle 12 value of 6.99, but still far in excess of the minimum 1.18 DNBR limit. Table 7.2.2–3 summarizes the sequence of events for the DNB hot full power CEA Withdrawal case.

In conclusion, the CEA Withdrawal event, when initiated from the Technical Specification LCOs, will not lead to a DNBR that violates the DNBR design limit. Furthermore, the initial available overpower margin requirements for this event are the most limiting for Cycle 16 at ASI conditions which are positive.

TABLE 7.2.2-1

FORT CALHOUN STATION UNIT NO. 1, CYCLE 16
KEY PARAMETERS ASSUMED IN THE HFP CEA WITHDRAWAL ANALYSIS

Parameter	Units	Cycle 16
Initial Core Power Level	MWt	1,500*
Core Inlet Coolant Temperature	°F	545*
Pressurizer Pressure	psia	2,075*
Moderator Temperature Coefficient	$\times 10^{-4}\ \Delta\rho/°F$	+0.5
Doppler Coefficient Multiplier		1.00
CEA Worth at Trip	$\%\Delta\rho$	6.061
Reactivity Insertion Rate Range	$\times 10^{-4}\ \Delta\rho/sec$	0 to 2.3
CEA Group Withdrawal Rate	in/min	46
Holding Coil Delay Time	sec	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.

TABLE 7.2.2-2

FORT CALHOUN STATION UNIT NO. 1, CYCLE 16
KEY PARAMETERS ASSUMED IN THE HZP CEA WITHDRAWAL ANALYSIS

Parameter	Units	Cycle 12	Cycle 16
Initial Core Power Level	MWt	1	1%*
Core Inlet Coolant Temperature	°F	532	532*
Pressurizer Pressure	psia	2,053	2,075*
Moderator Temperature Coefficient	x 10^{-4} $\Delta\rho$/°F	+0.5	+0.5
Doppler Coefficient Multiplier		0.85	1.00
CEA Worth at Trip	%$\Delta\rho$	5.28	6.789
Reactivity Insertion Rate Range	x 10^{-4} $\Delta\rho$/sec	0 to 1.0	0 to 2.9
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.

TABLE 7.2.2-3

FORT CALHOUN STATION UNIT NO.1, CYCLE 16
SEQUENCE OF EVENTS FOR THE HFP CEA WITHDRAWAL ANALYSIS

Time (sec)	Event	Setpoint or Value
0.000	Inadvertent Withdrawal of CEAs	------
672.0	Excore Power Approaches Trip Limit	112%
878.5	Time of Minimum DNBR	1.436

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 16 to determine the minimum initial overpower 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 to 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 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 16 analysis and compares them to the reference cycle (Cycle 12) values. Table 7.2.3-2 contains a sequence of events for the Loss of Flow analysis.

The low flow trip setpoint is reached at 2.54 seconds and the scram rods start dropping into the core 1.15 seconds later. A minimum CE-1 DNBR of 1.494 is reached at 4.4 seconds.

In conclusion, the Loss of Flow event ROPM requirements are bounded by the CEA withdrawal or Excess Load analysis for AOOs dependent upon initial available overpower margin. For Cycle 16 the Loss of Flow event, when initiated from the LCOs and in conjunction with the Low Flow Trip, will not exceed the minimum DNBR design limit.

TABLE 7.2.3-1

FORT CALHOUN STATION UNIT NO.1, CYCLE 16
KEY PARAMETERS ASSUMED IN THE LOSS OF COOLANT FLOW ANALYSIS

Parameter	Units	Cycle 12	Cycle 16
Initial Core Power Level	MWt	1,500*	1,500*
Core Inlet Coolant Temperature	°F	545*	545*
Initial RCS Flow Rate	gpm	208,280*	202,500*
Pressurizer Pressure	psia	2,075*	2,075*
Moderator Temperature Coefficient	$\times 10^{-4}\ \Delta\rho/°F$	+0.5	+0.5
Doppler Coefficient Multiplier		0.85	1.00
CEA Worth at Trip (ARO)	$\%\Delta\rho$	6.50	6.5086
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	1.77

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

TABLE 7.2.3-2

FORT CALHOUN STATION UNIT NO.1, CYCLE 16
SEQUENCE OF EVENTS FOR THE LOSS OF COOLANT FLOW ANALYSIS

Time (sec)	Event	Setpoint or Value
0.00	Loss of RCP	------
2.54	Low Flow Trip Setpoint Reached	93%
3.69	Scram Rods Drop into Core	------
4.4	Time of Minimum DNBR	1.494

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 16 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 16 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 100%, 90%, and 80% power with an ASI of −0.182, which is outside of the LCO limit of −0.08 at full power. This results in a minimum CE−1 DNBR of 1.401 (PDIL Case), and 1.44 (ARO Case). A maximum allowable initial linear heat generation rate of 15.5 kW/ft could exist as an initial condition without exceeding the acceptable fuel CTM limit of 22 kW/ft during this transient.

In conclusion, the CEA Drop event ROPM requirements are bounded by the CEA withdrawal at high powers, but will become limiting at lower powers, for AOOs 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 STATION UNIT NO. 1, CYCLE 16
KEY PARAMETERS ASSUMED IN THE HFP CEA DROP ANALYSIS

Parameter	Units	Cycle 11	Cycle 16
Initial Core Power Level	MWt	1,500*	1,500*
Core Inlet Coolant Temperature	°F	543*	545*
Core Mass Flow Rate	gpm	202,500*	202,500*
Pressurizer Pressure	psia	2,075*	2,075*
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	−0.2611
PDIL	$\%\Delta\rho$	−0.2295	−0.2619
Maximum Allowed Power Shape Index at Negative Extreme of LCO Band		−0.18	−0.18
Radial Peaking Distortion Factor			
Unrodded Region	$\%\Delta\rho$	1.1566	1.2079
Bank 4 Inserted	$\%\Delta\rho$	1.1598	1.2065

* 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 STATION UNIT NO.1, CYCLE 16
SEQUENCE OF EVENTS FOR FULL LENGTH CEA DROP HFP

Time (sec)	Event	Setpoint or Value
0.0	CEA Begins to Drop into Core	−−−−−−
1.0	CEA Reaches Fully Inserted Position	100% Insertion
1.2	Core Power Level Reaches a Minimum and Begins to Return to Power due to Reactivity Feedbacks	68% of 1500 MWt
83.0	Core Inlet Temperature Reaches a Minimum Value	539.39°F
200.0	RCS Pressure Reaches a Minimum Value	2,005.9 psia
200.0	RCS Power Returns to its Maximum Value	95.2% of 1,500 MWt
200.0	Minimum DNBR is Reached	1.44 (CE−1 Correlation ARO Condition)

7.2 ANTICIPATED OPERATIONAL OCCURRENCES (CATEGORY 2) (Continued)

7.2.5 Boron Dilution Event

The Boron Dilution event was reanalyzed for Cycle 16 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 16 and the reference cycle, Cycle 15. The Cycle 16 analysis utilized a mass basis in the calculations, as was used in Cycle 15, rather than a volumetric basis to ensure that all operating temperature ranges for all modes of operation were bounded.

Revisions to the Core Operating Limits Report for refueling boron concentration are necessary since the Cycle 16 value is greater than the Cycle 15 value. The boron dilution results for operator response times are shown in Table 7.2.5−2.

TABLE 7.2.5-1

FORT CALHOUN STATION UNIT NO. 1, CYCLE 16
KEY PARAMETERS ASSUMED IN THE BORON DILUTION ANALYSIS

Parameter	Cycle 15	Cycle 16
Critical Boron Concentration, ppm (ARO, No Xenon)		
Mode: Hot Standby	1,541	1,602
Hot Shutdown	1,541	1,602
Cold Shutdown – Normal RCS Volume	1,393	1,470
Cold Shutdown – Minimum RCS Volume*	1,213	1,262
Refueling	1,358	1,426
Inverse Boron Worth, ppm/%$\Delta\rho$		
Mode: Hot Standby	−55	−55
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)**	1,900	2,000***

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

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

*** Proposed Cycle 16 COLR value.

TABLE 7.2.5−2

FORT CALHOUN STATION UNIT NO. 1, CYCLE 16
TIME TO CRITICALITY FOR BORON DILUTION ANALYSIS

Mode of Operation	Time to Criticality (min.)	Acceptance Criteria (min.)
1. Hot Standby	N/A	N/A
2. Hot Standby	35.54	>15
3. Hot Shutdown	35.54	>15
Hot Shutdown to Cold Shutdown	36.25	>15
4. Cold Shutdown		
Undrained RCS	37.12	>15
Drained RCS	15.27	>15
5. Refueling Operations	42.04	>30

7.0 TRANSIENT ANALYSIS (Continued)

7.3 POSTULATED ACCIDENTS

7.3.1 CEA Ejection

The CEA Ejection event was not reanalyzed for Cycle 16 since the Westinghouse Cycle 14 analysis continues to bound input values from Cycle 16. A summary report was transmitted to the NRC for review in Reference 14.

7.3.2 Main Steam Line Break Accident

This accident was reanalyzed for Cycle 16 using the methodology discussed in References 6 and 12. The Main Steam Line Break (MSLB) accident was previously analyzed in the Fort Calhoun FSAR and satisfactory results were reported therein. The SLB accidents at both HZP and HFP were reanalyzed for Cycle 16 with acceptable results. The moderator cooldown curves for Cycle 16 are bounded by Cycle 8 as shown in Figure 7.3.2−1. Both the FSAR and reference cycle evaluations will be reported in the 1995 update of the Fort Calhoun Station Unit No. 1 USAR.

The MSLB event initiated from HFP was simulated using CESEC with parameters that maximize the potential for Return to Power (R−T−P) or/and Return to Criticality (R−T−C). The limiting MSLB accident occurs with all Reactor Coolant Pumps(RCPs) running. This case shows a peak R−T−P of 11.86%, a peak reactivity of −0.052458 %$\Delta\rho$ and a peak core heat flux of 11.57%. This is bounded by the Cycle 8 HFP RCPs operating case where there was a peak R−T−P of 12.92%. The Cycle 8 analysis results were bounded by the reference cycle (i.e., Cycle 1).

Other cases were run to confirm that the reactivity effects for the MSLB with the RCPs tripped after SIAS are less severe than all RCPs operating. Two Loss of AC cases were run: (1) AC power loss at time of break, and (2) AC loss at time of trip.

It has been determined that the MSLB case with RCPs tripped is similar to the MSLB case with a loss of offsite power since the RCPs coast down in both events. The consequences of the MSLB event with the Trip2/Leave2 strategy (current Emergency Operating Procedures) is bounded by the loss of offsite power. The Cycle 16 events are less severe than those analyzed for Cycles 1 and 8 and continue to be bounded by the reference cycle.

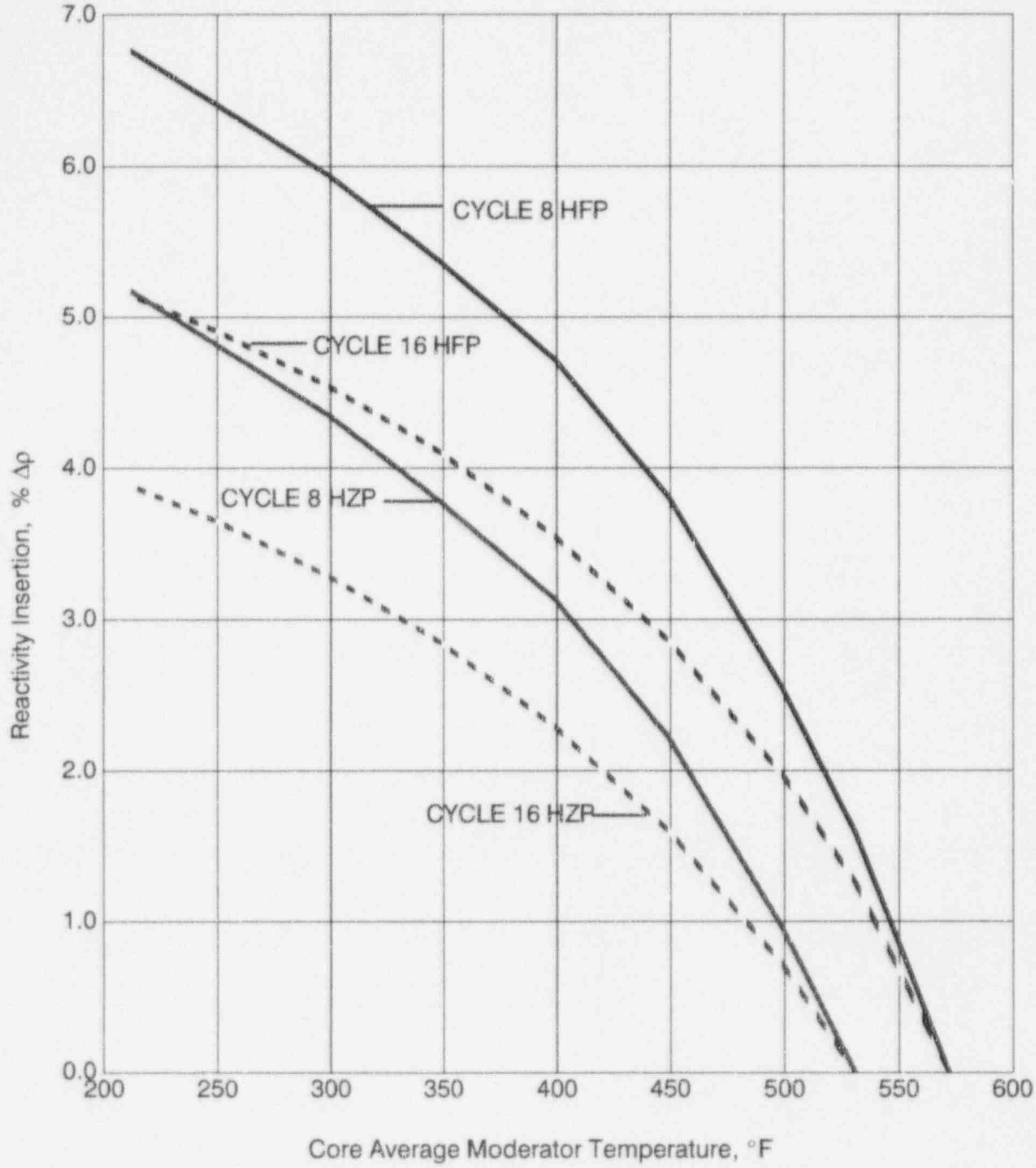
7.0 TRANSIENT ANALYSIS (Continued)

7.3 POSTULATED ACCIDENTS (Continued)

7.3.2 Main Steam Line Break Accident (Continued)

The most limiting Cycle 16 HZP case is with RCPs operating and when the Technical Specification (TS) limit of 4.0 $\%\Delta\rho$ shutdown margin is conservatively used for the scram curve multiplier. This methodology was used in the Cycle 8 HZP analysis. The methodology conservatively assumes that at the HZP condition the minimum CEA worth available for negative reactivity addition at time of trip will be equivalent to the minimum allowable shutdown margin of TS Section 2.10.2(1). The TS reactivity control limits require that whenever the reactor is in hot standby or power operation condition with $T_{cold} > 210°F$, a shutdown margin of $\geq 4.0\ \%\Delta\rho$ must be available. In actuality, the minimum available scram worth with most reactive rod stuck out calculated for the MSLB is 5.082 $\%\Delta\rho$ for HZP/BOC (PDIL). This minimum value is considerably greater than the 4.0 $\%\Delta\rho$ TS minimum shutdown margin allowed.

The limiting HZP case shows a peak R–T–P of 0.08%, a peak reactivity of $-0.34161\ \%\Delta\rho$, and a peak core heat flux of 4.09%. This is bounded by the Cycle 8 HZP case where there was a peak R–T–P of 19.20% and a peak reactivity of $+0.353\ \%\Delta\rho$. The Cycle 1 (reference cycle) analysis results are more limiting than those of Cycle 8. Thus, the Cycle 16 results continue to be bounded by the reference cycle.



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

7.3.3 Seized Rotor Event

The Seized Rotor event was reviewed for Cycle 16 to demonstrate that only a small fraction of fuel pins are predicted to fail during this event. The analysis showed that Cycle 16 is bounded by the reference cycle (Cycle 9) analysis because: (1) an F_R^T TS limit of 1.85 was assumed in the Cycle 9 analysis compared to the Cycle 16 F_R^T COLR limit of 1.77, and (2) the failed fuel pin percentage during a Cycle 16 seized rotor event was calculated to be approximately 0.05165 %. This is far below the 1% pin failure threshold above which dose rate calculations are required to demonstrate that the 10 CFR 100 limits are not exceeded.

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 the Large and Small Break Loss of Coolant Accident (LOCA) evaluations, in accordance with 10 CFR 50.46, Appendix K, 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 used in evaluating the non–LOCA transients to ensure the fuel mechanical design requirements were valid for the operation of Cycle 16.

Westinghouse has notified OPPD that the ECCS analysis had a significant peak clad temperature (PCT) change in 1994. The results of the revised Small Break LOCA evaluation will be included in the 1995 USAR revision. No code errors have been reported during Cycle 15 which significantly change the PCT for the Large Break LOCA (Reference 15). The results of the Small and Large Break LOCA evaluations have been confirmed to be applicable to the proposed Cycle 16 operation.

9.0 STARTUP TESTING

The startup testing program proposed for Cycle 16 is identical to that used in Cycle 15. 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 will be substituted for the full core symmetry checks. The pseudo–eject symmetry check test was eliminated as described in References 3 and 4.

The CEA exchange technique is a method for measuring rod worths which is faster and produces less waste than the typical boration/dilution method. The startup testing method used in Cycles 11 through 15 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 low power CECOR maps provide for a less time consuming but equally valid technique for detecting azimuthal power tilts during reload core physics testing.

The acceptance and review criteria for these tests are:

Test	Acceptance Criteria	Review Criteria
CEA Group Worths	$\pm$ 15% of predicted	$\pm$ 15% of predicted
Low Power CECOR Maps	Technical Specification limits of F_R^T, F_{XY}^T, 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 06, May 1994.
2. "Omaha Public Power District Reload Core Analysis Methodology – Neutronics Design Methods and Verification," OPPD–NA–8302–P, Revision 04, May 1994.
3. "Omaha Public Power District Reload Core Analysis Methodology – Transient and Accident Methods and Verification," OPPD–NA–8303–P, Revision 04, January 1993.
4. "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−162−P−A, September 1976.
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 06, May 1994.
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. Letter LIC-94-0142, W. G. Gates (OPPD) to U. S. Nuclear Regulatory Commission (Document Control Desk), Docket No. 50-285, "Safety Analysis Report Update and 10CFR50.59 Report for Fort Calhoun Station," dated July 1, 1994.
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, and 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 04, January 1993.
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, Revision 04, May 1994.
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.

10.0 REFERENCES (Continued)

References (Chapter 8)

1. "Omaha Public Power District Reload Core Analysis Methodology – Transient and Accident Methods and Verification," OPPD–NA–8303–P, Revision 04, January 1993.
2. Letter LIC–91–247R, W. G. Gates (OPPD) to Document Control Desk (NRC), dated September 30, 1991.

Referencec (Chapter 9)

1. "Control Rod Group Exchange Technique," CEN–319, November 1985.
2. "Acceptance for Referencing of Licensing Topical Report CEN–319 – Control Rod Group Exchange Technique," letter, Dennis M. Crutchfield (NRC) to Rik W. Wells (Chairman – CE Owners Group), dated April 16, 1986.
3. Letter LIC–93–0254, W. G. Gates (OPPD) to Document Control Desk (NRC), dated October 1, 1993.
4. Letter S. Bloom (NRC) to T. L. Patterson (OPPD), dated November 10, 1993.