OCONEE UNIT 3, CYCLE 12

- Reload Report -

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CONTENTS

List of Tables

List of Figures

1. INTRODUCTION AND SUMMARY

This report justifies the operation of the twelfth cycle of Oconee Nuclear Station, Unit 3, at the rated core power of 2568 MWt. Included are the required analyses as outlined in the USNRC document "Guidance for Proposed License Amendments Relating to Refueling", June 1975.

To support Cycle 12 operation of Oconee Unit 3, this report employs analytical techniques and design bases established in reports that were previously submitted and accepted by the USNRC and its predecessor (see references).

A brief summary of Cycle 11 and 12 reactor parameters related to power capability is included in Section 5 of this report. All of the accidents analyzed in the FSAR[1] have been reviewed for Cycle 12 operation. In those cases where Cycle 12 characteristics were conservative compared to those analyzed for previous cycles, no new accident analyses were performed.

The Technical Specifications have been reviewed, and the modifications, if any, required for Cycle 12 operation are justified in this report.

Based on the analyses performed, which take into account the postulated effects of fuel densification and the Final Acceptance Criteria for Emergency Core Cooling Systems, it has been concluded that Oconee Unit 3 can be operated safely for Cycle 12 at the rated power level of 2568 MWt.

2. OPERATING HISTORY

The referenced fuel cycle for the nuclear and thermal-hydraulic analyses of Oconee Unit 3, Cycle 12, is the currently operating Cycle 11. Cycle 11 achieved initial criticality on September 22, 1988 and power escalation commenced on September 23, 1988. The fuel cycle design length for Cycle 12 - 410 $\pm$ 10 EFPD - is based on a Cycle 11 length of 410 $\pm$ 10 EFPD. No operating anomalies occurred during previous cycle operations that would adversely affect fuel performance in Cycle 12.

Cycle 12 will operate in a feed-and-bleed mode for its entire design length, as did Cycle 11.

3. GENERAL DESCRIPTION

The Oconee Unit 3 reactor core and fuel design basis are described in detail in Chapter 4, of the FSAR[1]. The Cycle 12 core consists of 177 fuel assemblies, each of which is a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The fuel consists of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The fuel assemblies in all batches have an average nominal fuel loading of 463.6 kg uranium. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in Table 4-1.

Figure 3-1 is the core loading diagram for Oconee 3, Cycle 12. All 16 Batch 8 assemblies will be discharged at the end of Cycle 11 along with 12 of the Batch 9 assemblies, 21 of the Batch 11 assemblies, and 3 of the Batch 12 assemblies. The remaining 4 Batch 9 assemblies, designated "9D", have an initial enrichment of 3.18 wt% ^{235}U. The remaining 4 Batch 11 assemblies and 57 Batch 12 assemblies, designated "11C" and "12B", respectively, all have an initial enrichment of 3.22 wt% ^{235}U. The core periphery is composed of the Batch 9 and Batch 11 assemblies, along with some of the Batch 12 and Batch 13 (3.48 wt% ^{235}U) assemblies. The Batch 14 assemblies, with an initial enrichment of 3.80 wt% ^{235}U, are distributed relatively evenly throughout the core interior with the rest of the Batch 12 and Batch 13 assemblies. Figure 3-2 is a quarter-core map showing the assembly burnup and enrichment distribution at the beginning of Cycle 12.

Cycle 12 will operate in a rods-out, feed-and-bleed mode. Core reactivity control is supplied mainly by soluble boron and supplemented by 61 full-length Ag-In-Cd control rods and 44 burnable poison rod assemblies (BPRAs). In addition to the full-length

control rods, eight Inconel gray axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The Cycle 12 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The Cycle 12 locations and enrichments of the BPRAs are shown in Figure 3-4.

FIGURE 3.1. CORE LOADING DIAGRAM FOR OCONEE 3, CYCLE 12

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						O11 12B	G10 13	F10 11C	G06 13	O05 12B					
B				P10 12B	N11 13	D13 13	14	F05 13	14	D03 13	N05 13	P06 12B			
C			L15 9D	14	G12 13	14	P11 12B	B07 13	P05 12B	14	G04 13	14	R06 9D		
D		L14 12B	14	H07 12B	14	K01 12B	14	E08 12B	14	K15 12B	14	G08 12B	14	L02 12B	
E		M12 13	N07 13	14	D04 12B	14	K03 12B	14	K13 12B	14	D12 12B	14	N09 13	M04 13	
F	M13 12B	O04 13	14	A09 12B	14	H15 12B	L03 13	C08 13	L13 13	R08 12B	14	A07 12B	14	O12 13	M03 12B
G	L07 13	14	M14 12B	14	C09 12B	C10 13	E06 13	B09 13	F11 13	C06 13	C07 12B	14	M02 12B	14	L09 13
HW	F06 11C	M06 13	K02 13	H05 12B	14	H03 13	G02 13	K09 12B	K14 13	H13 13	14	H11 12B	G14 13	E10 13	L10 11C
K	F07 13	14	E14 12B	14	O09 12B	O10 13	L05 13	P07 13	M10 13	O06 13	O07 12B	14	E02 12B	14	F09 13
L	E13 12B	C04 13	14	R09 12B	14	A08 12B	F03 13	O08 13	F13 13	H01 12B	14	R07 12B	14	C12 13	E03 12B
M		E12 13	D07 13	14	N04 12B	14	G03 12B	14	G13 12B	14	N12 12B	14	D09 13	E04 13	
N		F14 12B	14	K08 12B	14	G01 12B	14	M08 12B	14	G15 12B	14	H09 12B	14	F02 12B	
O			A10 9D	14	K12 13	14	B11 12B	P09 13	B05 12B	14	K04 13	14	F01 9D		
P				B10 12B	D11 13	N13 13	14	L11 13	14	N03 13	D05 13	B06 12B			
R						C11 12B	K10 13	L06 11C	K06 13	C05 12B					

Y

Z

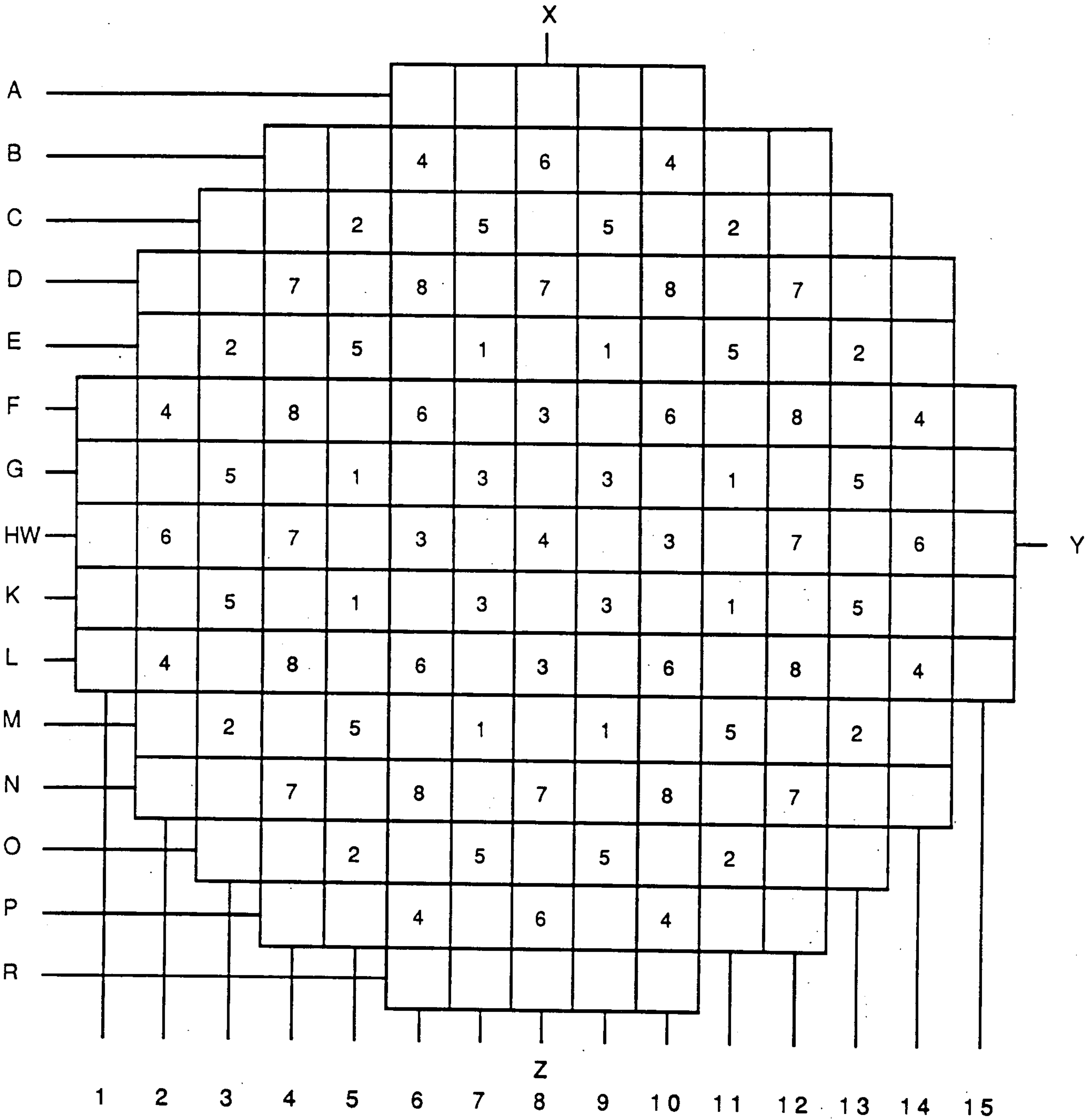
XX X	PREVIOUS CYCLE LOCATION BATCH NO.

FIGURE 3.2 ENRICHMENT & BURNUP FOR OCONEE 3, CYCLE 12

	8	9	10	11	12	13	14	15
H	3.22 32543	3.48 14654	3.48 17456	3.80 0	3.22 28597	3.48 14644	3.48 18114	3.22 35494
K	3.48 14654	3.48 18063	3.48 17017	3.22 33375	3.80 0	3.22 25084	3.80 0	3.48 17556
L	3.48 17446	3.48 16999	3.22 24136	3.80 0	3.22 24055	3.80 0	3.48 12857	3.22 28873
M	3.80 0	3.22 33361	3.80 0	3.22 31759	3.80 0	3.48 17833	3.48 17745	
N	3.22 28580	3.80 0	3.22 24094	3.80 0	3.22 28952	3.80 0	3.22 27678	
O	3.48 14644	3.22 25124	3.80 0	3.48 17812	3.80 0	3.18 33897		
P	3.48 18114	3.80 0	3.48 12849	3.48 17713	3.22 27685			
R	3.22 35494	3.48 17533	3.22 28863					

X.XX XXXXX	INITIAL ENRICHMENT, wt% ^{235}U BOC BURNUP, MWd/mtU

FIGURE 3.3. CONTROL ROD LOCATIONS FOR OCONEE 3, CYCLE 12



X GROUP NO.

GROUP	NO. OF RODS	FUNCTION
1	8	SAFETY
2	8	SAFETY
3	8	SAFETY
4	9	SAFETY
5	12	CONTROL
6	8	CONTROL
7	8	CONTROL
8	8	APSRs
	TOTAL 69	

FIGURE 3.4 BPRA ENRICHMENT & DISTRIBUTION FOR OCONEE 3, CYCLE 12

	8	9	10	11	12	13	14	15
H				1.0				
K					1.1		0.0	
L				1.0		1.1		
M	1.0		1.0		1.1			
N		1.1		1.1				
O			1.1					
P		0.0						
R								

XX

BPRA CONCENTRATION, wt% B_4C IN Al_2O_3

4. FUEL SYSTEM DESIGN

4.1 Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters for Oconee 3 Cycle 12, are listed in Table 4-1. All fuel assemblies are identical in concept and are mechanically interchangeable. Two regenerative neutron sources will be used in Mark B7 fuel assemblies.

The Batch 14 MK-B8 fuel assemblies are similar in design to the MK-B7 fuel previously reloaded into Oconee 3 Cycle 11 (Reference 3). Like the MK-B7 fuel design, the MK-B8 features intermediate Zircaloy grids, a skirtless inconnel upper end grid (dimensionally equivalent to the intermediate Zircaloy grids), and a removable upper end fitting. The new features for the MK-B8 is a debris fretting resistant fuel rod design, which utilizes a lengthened solid lower end plug extending below the bottom end grid. The fuel rod's prepressurization level was also reduced slightly to compensate for the reduction in plenum void volume.

The Oconee 3 Cycle 12 core will have 44 BPRAs inserted in the Batch 14 fuel assemblies. Thirty-six (36) of the BPRAs will be new, and the remaining 8 will be reinserted from the Cycle 11 core (once burned).

Other results presented in the FSAR[1] fuel assembly mechanical discussions and in previous reload reports, are applicable to the reload fuel assemblies. Duke has performed generic mechanical analyses, as described below, which envelope the Cycle 12 design. All methods are consistent with the approved methodologies of Reference 8 except where specifically stated.

4.2 Fuel Rod Design

The mechanical evaluation of the fuel rod is discussed below.

4.2.1 Cladding Collapse

The fuel of Batches 9D, 11C, and 12B are more limiting than other batches due to their longer previous incore exposure time. These assembly power histories were assessed against Duke's creep collapse analyses, which are based on the CROV computer code and procedures described in topical report BAW-10084, Rev. 2[11]. The TACO2[4] code was used to calculate internal pin pressures and clad temperatures used as input to CROV. The collapse time for the most limiting assembly was conservatively determined to be > 42,000 EFPH, which is greater than the maximum projected residence time of Cycle 9D fuel (Table 4-1).

4.2.2 Cladding Stress

As described in Reference 8, Duke has performed a generic and conservative fuel rod cladding stress analysis in accordance with the guidelines set forth in Section III, Division 1 - Subsection NB, of the ASME Boiler and Pressure Vessel Code. All methods are consistent with Reference 8. Compliance with ASME Code criteria verify the structural integrity of the cladding throughout the most limiting design conditions.

The following conservatisms exist in the generic cladding stress calculation:

- high external cladding pressure (110% of design system pressure)
- low internal pressure (HZP - min. specified pre-pressure)
- maximum possible radial temperature gradient through clad (fuel melt conditions)
- conservative cladding dimensions with regard to stress

4.2.3 Cladding Strain

Duke has performed a cladding strain calculation using TACO2[4] in

accordance with the approved methodology[8]. This analysis demonstrated that the uniform, circumferential strain of the cladding was within 1.0%.

4.3 Thermal Design

All fuel assemblies in the Cycle 12 core are thermally similar. The fresh Batch 14 fuel inserted for Cycle 12 operation introduces no significant differences in fuel thermal performance relative to the other fuel remaining in the core. The linear heat rate to melt capability based on centerline fuel melt was assessed separately for each batch of fuel against Duke's generic linear heat rate to melt analysis, which is based on the TACO2[4] computer code. The fuel parameters used to determine the fuel melt limits are shown in Table 4-2.

The analysis includes the following bounding, generic conservatisms:

1. A maximum gap based on as-fabricated pellet and cladding data.

2. Maximum incore densification based on resinter test results.

The burnup dependent linear heat rate (LHR) capability and the average fuel temperature for each batch are shown in Table 4-2.

The maximum assembly average burnup is predicted to be 45948 MWD/MTU and the maximum fuel rod burnup is predicted to be 47828 MWD/MTU. Fuel rod internal pressure has been evaluated using TACO2[4] with a conservative pin power history, and the maximum pressure is less than the nominal reactor coolant (RC) system pressure of 2200 psia.

4.4 Material Design

The Batch 14 fuel assemblies are not unique in concept, nor do they utilize different component materials. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the Batch 14 fuel assemblies is identical to those of the present fuel.

Table 4-1

Fuel Design Parameters and Dimensions

	Batch No.				
	9D	11C	12B	13	14
FA type	Mark B5	Mark B5Z	Mark B5Z	Mark B7	Mark B8
No. of FAs	4	4	57	60	52
Fuel rod OD, in.	0.430	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377	0.377
Flex spacers, type	Spring	Spring	Spring	Spring	Spring
Rigid spacers, type	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Undensified active fuel length, in.	141.8	141.8	141.8	141.8	141.8
Fuel pellet OD (mean spec), in.	0.3686	0.3686	0.3686	0.3686	0.3686
Fuel pellet initial density (mean spec), %TD	95.0	95.0	95.0	95.0	95.0
Initial fuel enrichment, wt % ^{235}U	3.18	3.22	3.22	3.48	3.80
Est. residence time, EOC 12, Hours	40,224	39,048	30,672	19,920	10,080
Cladding collapse time, Hours	>42,000	>42,000	>34,000	>25,000	>15,000

Table 4-2. Linear Heat Rate to Melt Analysis

	Batch No.				
	9D	11C	12B	13	14
Nominal initial density, % TD	95.0	95.0	95.0	95.0	95.0
Nominal initial pellet diameter, in.	0.3686	0.3686	0.3686	0.3686	0.3686
Nominal initial clad ID, in.	0.377	0.377	0.377	0.377	0.377
Nominal initial clad OD, in.	0.430	0.430	0.430	0.430	0.430
Average linear heat rate @ 100% of 2568 MW, kW/ft	5.74	5.74	5.74	5.74	5.74
Linear heat rate capability[b] from 0-1000 MWD/MTU, kW/ft	20.15	20.15	20.15	20.15	20.15
Linear heat rate capability[b] >1000 MWD/MTU, kW/ft	21.20	21.20	21.20	21.20	21.20
Average fuel temp. @ nominal linear heat rate, °F	1240[a]	1240[a]	1240[a]	1240[a]	1240[a]

(a) Basis: TACO2[4], 96.5% TD @ 4000 MWD/MTU, nominal pellet and cladding dimensions.

(b) These values are utilized as fuel design limits for Cycle 12.

5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 compares the core physics parameters of design Cycle 12 with those of the reference Cycle 11. The Cycle 11 and 12 values were generated by Duke Power Company using the CASMO-2 based reload design methods described in Reference 8. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles. Figure 5-1 illustrates a representative relative power distribution for the beginning of Cycle 12 at full power with equilibrium xenon and normal rod positions.

The primary reasons for the differences in the physics parameters between Cycles 11 and 12 are the variation in the shuffle pattern and the use of a 52 feed core for Cycle 12. Differences in ejected and stuck rod worths between cycles are due to changes in the radial flux and burnup distributions. All safety criteria associated with these rod worths are met. The adequacy of the shutdown margin with Cycle 12 stuck rod worths is demonstrated in Table 5-2. The following conservatisms were applied for the shutdown calculations:

1. Poison material depletion allowance.
2. 10% uncertainty on net rod worth.

Flux redistribution was explicitly accounted for since the shutdown analysis was calculated using a three-dimensional model. The reference fuel cycle shutdown margin is presented in the Oconee 3, Cycle 11 Reload Report[3].

5.2 Analytical Input

The Cycle 12 incore measurement calculation constants to be used to compute core power distributions were obtained in a similar manner for Cycle 12 as for the reference cycle.

5.3 Changes in Nuclear Design

The methodology described in Reference 8 has been implemented for Oconee 3 Cycle 12.

Table 5-1. Oconee 3 Physics Parameters[a]

	Cycle 11[b]	Cycle 12[c]
Cycle length, EFPD	410	410
Cycle burnup, MWd/mtU	12831	12,831
Average core burnup, EOC, MWd/mtU	27169	29169
Initial core loading, mtU	82.1	82.1
Critical boron - BOC (no xenon), ppm		
HZP, groups 7 and 8 at nominal positions[d]	1635	1662
HFP, groups 7 and 8 at nominal positions	1408	1425
Critical boron - EOC (equilibrium xenon), ppm		
HZP, groups 7 and 8 at nominal positions	391	370
HFP, groups 7 and 8 at nominal positions	5	9
Control rod worths - HFP, BOC, % dk/k		
Group 7	0.99	0.90
Group 8[e]	0.16	0.15
Control rod worths - HFP, EOC, % dk/k		
Group 7	1.06	1.04
Group 8[e]	(F)	(F)
Max ejected rod worth - HZP, % dk/k		
BOC, groups 5-8 inserted	0.36(L10)	0.25(L10)
EOC, groups 5-8 inserted	0.41(L10)	0.34(N12)
Max stuck rod worth - HZP, % dk/k		
BOC	1.37(N12)	1.05(M13)
EOC	1.54(N12)	1.41(M13)
Power deficit, HFP to HZP, % dk/k		
BOC	1.87	1.89
EOC	3.21	3.16
Doppler coeff - HFP, 10^{-5} (dk/k-°F)		
BOC (equilibrium xenon)	-1.20	-1.23
EOC (equilibrium xenon)	-1.54	-1.56

Table 5-1. (cont'd)

	Cycle 11[b]	Cycle 12[c]
Moderator coeff - HFP, 10^{-4} (dk/k-°F)		
BOC (no xenon)	-0.93	-1.00
EOC(equilibrium xenon)	-3.40	-3.42
Boron worth - HFP, ppm/% dk/k		
BOC	127	129
EOC	115	116
Xenon worth - HFP, % dk/k		
BOC (4 days)	2.59	2.56
EOC (equilibrium)	2.74	2.71
Effective delayed neutron fraction - HFP		
BOC	0.00609	0.00600
EOC	0.00515	0.00517

(a) Cycle 12 data are for the conditions stated in this report. The Cycle 11 core conditions are identified in Reference 3.

(b) Based on a 440 ± 10 EFPD Cycle 10 (Actual Cycle 10 length of 448.25 EFPD).

(c) Based on a Cycle 11 length of 410 ± 10 EFPD.

(d) Nominal positions are as follows:

	Cycle 11	Cycle 12
HZP (BOC)	CRGP7,8 = 100,35% WD	CRGP7,8 = 100,35% WD
HZP (EOC)	CRGP7,8 = 100,100% WD	CRGP7,8 = 100,100% WD
HFP (BOC)	CRGP7,8 = 92,35% WD	CRGP7,8 = 92,35% WD
HFP (EOC)	CRGP7,8 = 92,100% WD	CRGP7,8 = 92,100% WD

(e) Worth is calculated from 35% to 100% WD for Cycles 11 and 12.

(f) CRGP8 = 100% WD, therefore, there is no CRGP8 worth at EOC.

Table 5-2. Shutdown Margin Calculation for Oconee 3, Cycle 12

	BOC, % dk/k	EOC, % dk/k
Available Rod Worth		
Total rod worth, HZP	7.99	8.54
Worth reduction due to poison burnup	-0.42	-0.42
Maximum stuck rod, HZP	-1.05	-1.41
Net worth	6.52	6.71
Less 10% uncertainty	-0.65	-0.67
Total available worth	5.87	6.04
Required Rod Worth		
Power deficit, HFP to HZP	1.89	3.16
Max inserted rod worth, HFP	0.32	0.49
Total required worth	2.21	3.65
Shutdown Margin		
Total available worth minus total required worth	3.66	2.39

Note: Required shutdown margin is 1.00% dk/k.

FIGURE 5-1
OCONEE 3, CYCLE 12
TWO DIMENSIONAL
RELATIVE POWER DISTRIBUTION

HFP, 004 EFPD, EQXE
NOMINAL ROD POSITIONS

	8	9	10	11	12	13	14	15
H	0.976	1.274	1.262	1.372	1.048	1.175	1.075	0.398
K	1.274	1.246	1.184	0.965	1.352	1.089	1.292	0.529
L	1.262	1.184	1.079	1.341	1.072	1.348	1.002	0.316
M	1.372	1.965	1.341	0.997	1.313	1.078	0.633	
N	1.048	1.352	1.072	1.313	0.924	1.061	0.331	
O	1.175	1.089	1.348	1.078	1.061	0.347		
P	1.075	1.292	1.002	0.633	0.331			
R	0.398	0.529	0.316					

6. THERMAL-HYDRAULIC DESIGN

The generic Mark-B and Mark-BZ thermal-hydraulic design analyses supporting Cycle 12 operation were performed by Duke Power Company using the methods described in References 1, 2, 3, 6, and 8. The Cycle 11 and Cycle 12 maximum design conditions are summarized in Table 6-1.

The Cycle 12 transition core will include 52 fresh Mark-B8 Batch 14 fuel assemblies, all but eight of which contain BPRAs. Two assemblies will contain regenerative neutron sources, leaving 62 fuel assemblies with open guide tubes. This results in a core bypass flow of 8.6% of the total system flow. This bypass flow is less than that assumed in the generic thermal-hydraulic analyses and the consequent increase in core flow establishes the generic analysis as conservative for Cycle 12 operation.

The Mark-BZ fuel assembly has a slightly higher pressure drop than the Mark-B assembly as a result of the increased flow resistance of the Zircaloy spacer grids. The Cycle 12 transition core was conservatively analyzed at the limiting thermal design conditions using VIPRE-01 and the MDNBR is greater than the BWC CHF correlation limit of 1.18, Reference 9.

In a Mark-BZ transition core the limiting Mark-B hot channel will receive more coolant flow and yield better DNB performance than would be predicted for a full Mark-B core. Thus, the generic Mark-B analyses, based on the B&W-2 CHF correlation, are bounding and are applicable to the Cycle 12 transition core.

No fuel rod bow penalty was included in the DNBR limit used in the generic Mark-BZ analyses, as justified in Reference 7. The rod bow topical report concludes that a DNBR penalty is no longer required for thermal-hydraulic analyses. Nevertheless, to account for fuel rod bow, the generic Mark-B analyses used for determining plant

operating limits (except the flux to flow setpoint analysis) were based on a DNBR criteria including 10.2% margin from the 1.30 design limit. Primarily due to this conservatism, the current pressure-temperature envelope and design radial x local peaking have been shown to be conservative for a full and transition Mark-BZ core.

A flux to flow setpoint of 1.07 will be used for Cycle 12 operation. A conservative two pump coastdown analysis was performed based on a 1.07 flux to flow setpoint and the reference design radial-local peaking factor, F_{dH} = 1.714. The two pump coastdown is initiated from a power level of 102% full power. This initial condition is based on 100% indicated power plus a 2% allowance for the secondary side heat balance uncertainty. The reactor trip time was conservatively determined based on an NI indicated power of 98%, 100% minus 2% NI calibration error. The minimum DNBR determined in the transition core flux to flow analysis is greater than the BWC CHF correlation limit of 1.18, Reference 9. The minimum DNBR determined in the generic Mark-B flux to flow analysis, conservatively based on a 1.07 flux to flow setpoint, is greater than the BAW-2 CHF correlation limit of 1.30.

Table 6-1. Thermal Hydraulic Design Conditions

	Cycle 11	Cycle 12
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design flow	109.5	108.5
Core bypass flow[a], % total flow	7.9	8.6
Vessel inlet/outlet coolant temp at 100% power, °F	556.2/601.8	554.7/603.4
Ref design radial-local power peaking factor	1.71	1.71
Ref design axial flux shape	1.5 cosine	1.5 cosine
Active fuel length, in.	141.8	141.8
Avg heat flux at 100% power, 10^3 Btu/hr-ft^2	176[b]	174
CHF correlation	BAW-2/BWC	BAW-2/BWC
Min DNBR with densification penalty	>1.30/>1.18	>1.30/>1.18
Hot channel factors: Enthalpy rise	1.011/1.011	1.01/1.0111
Heat flux	1.014/1.014	1.01/1.0144
Flow area	0.98/0.97	0.98/0.97

(a) Generic analyses based on ≥8.0% core bypass flow for cycle 11, ≥9.0% core bypass flow for Cycle 12.

(b) Heat flux based on a conservative minimum densified length of 140.3 in.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in Cycle 11 parameters to determine the effect of the Cycle 12 reload and to ensure that thermal performance during hypothetical transients is not degraded. The effects of fuel densification on the FSAR accident results have been evaluated and are reported in Reference 6. Since Batch 14 reload fuel assemblies contain fuel rods with a theoretical density higher than those considered in Reference 6, the conclusions in that reference are still valid.

No new dose calculations were performed for the reload report. The dose considerations in Reference 15 are characteristic for Oconee 3 Cycle 12 based upon comparisons of key parameters which determine radionuclide inventories.

7.2 Accident Evaluations

The key parameters that have the greatest effect on determining the outcome of a transient can typically be classified in three major areas: core thermal parameters, thermal-hydraulic parameters, and kinetics parameters, including the reactivity feedback coefficients and control rod worths.

Fuel thermal analysis parameters for each batch in Cycle 12 are given in Table 4-2. Table 6-1 compares the Cycle 11 and 12 thermal-hydraulic maximum design conditions. Table 7-1 compares the key kinetics parameters from the FSAR and Cycle 12. The effect of a more negative hot full power end-of-cycle moderator temperature coefficient on the FSAR accident analyses has been analyzed for Oconee Nuclear Station[13]. Table 7-1 has been revised to include the new values for end-of-cycle moderator temperature coefficient and dropped rod worth assumed in these analyses.

A generic LOCA analysis for the B&W 177-FA, lowered-loop NSSS has been performed using the Final Acceptance Criteria ECCS Evaluation Model given in BAW-10103, Rev. 3[10]. The LOCA kW/ft limits given in BAW-10103 have been impacted by TACO2, NUREG-0630, and FLECSET. The net effect of these factors is summarized by the kW/ft limits in BAW-1915[14].

The combination of average fuel temperature as a function of LHR and the lifetime pin pressure data used in the BAW-1915 LOCA limits analysis[14] is conservative compared to those calculated for this reload. In addition, it has been determined that the slightly lower prepressurization of the Batch 14 fuel rods does not have an adverse impact on the LOCA analyses[12]. Thus, the analysis and the LOCA limits reported in BAW-1915 provide conservative results for the operation of Oconee 3 Cycle 12 fuel[5].

The LOCA kW/ft limits have been reduced for the first 25 EFPDs. The kW/ft limits for the first 25 EFPDs are shown in Table 7-2. Table 7-3 shows the bounding values for allowable LOCA peak LHRs for Oconee 3 Cycle 12 fuel after 25 EFPD.

From the examination of Cycle 12 core thermal properties and kinetics properties with respect to acceptable previous cycle values, it is concluded that this core reload will not adversely affect the safe operation of Oconee 3 during Cycle 12. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of Cycle 12 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in Cycle 12 are bounded by the FSAR and/or the fuel densification report[6].

Table 7.1. Comparison of Key Parameters for Accident Analysis

Parameter	FSAR[1] value	Predicted Cycle 12 value
BOC Doppler coeff, HFP, 10^{-5}, dk/k/°F	-1.17	-1.23
EOC Doppler coeff, HFP, 10^{-5}, dk/k/°F	-1.33[a]	-1.56
BOC moderator coeff, HFP, 10^{-4},dk/k/°F	+0.5[b]	-1.00
EOC moderator coeff, HZP, 10^{-4},dk/k/°F	-3.0[c]	-2.93
EOC moderator coeff, HFP, 10^{-4},dk/k/°F	-3.5[c]	-3.42
All rod bank worth, HZP, % dk/k	10.0	8.54
Boron reactivity worth, 70°F ppm/1% dk/k	75	94
Max. ejected rod worth, HFP, % dk/k	0.65	0.22
Dropped rod worth, HFP, % dk/k	0.40	0.12
Initial boron conc, HFP, ppm	1400	1425[d]

(a) -1.2×10^{-5} dk/k/°F was used for steam line break analysis. -1.3×10^{-5} dk/k/°F was used for cold water accident (pump start-up).

(b) $+0.94 \times 10^{-4}$ dk/k/°F was used for the moderator dilution accident.

(c) The HZP moderator temperature coefficient is one of the key parameters assumed in the steam line break analysis. The HFP moderator temperature coefficient is included since it is one of the parameters assumed in the cold water, rod ejection, and control rod misalignment accident analyses, although none of these accidents are very sensitive to changes in the coefficient.

(d) The combined effect of boron concentration and boron worth is conservative for Cycle 12.

Table 7-2. LOCA Limits, Oconee 3, Cycle 12 0-1000 Mwd/mtU[a]

Elevation ft	LHR Limits, kW/ft
2	14.0
4	16.1
6	16.5
8	17.0
10	16.0

Table 7-3. LOCA Limits, Oconee 3, Cycle 12, After 1000 Mwd/mtU

Elevation, ft	LHR limits, kW/ft
2	15.5
4	16.6
6	18.0
8	17.0
10	16.0

[a]1000 MWd/mtU corresponds to approximately 25 EFPD for the most limiting assembly.

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for Cycle 12 operation in accordance with the methods of References 2 and 8 to account for minor changes in power peaking and control rod worths.

The figures for the operating limits on rod index and axial power imbalance have been removed from Section 3 of the Technical Specifications and will be included in the cycle specific Core Operational Limits Report (COLR). Both the operational power imbalance limits and rod index limits are revised and included in the COLR for Cycle 12.

The core protection safety limit setpoints are revised for O3C12. Figure 8-1 shows the revised safety limits. The imbalance values are error adjusted in the same manner as done in the past. The power values are error adjusted to account for a 2% heat balance error, a 2% Tech Spec allowance for calibration of the excore detectors to the heat balance, a 2% transient NI error, and an allowance for the uncertainty of the flux/flow imbalance trip function hardware. These uncertainty allowances are combined statistically using the square root sum of the square methodology to arrive at an overall 5% allowance for the power uncertainty. Error adjustment of the flux/flow/imbalance safety limits falls outside the current Reactor Protective System (RPS) maximum allowable setpoints, thus making it unnecessary to revise the Tech Specs for the RPS setpoints.

It has been proposed that all specifications related to two RCP operation be removed from the Technical Specifications. Figure 8-1 reflects this change, showing only the setpoints for 3 and 4 RCP operation. Figure 8-2 shows the new Tech Spec figure for the RPS maximum allowable setpoints which excludes the 2 RCP setpoints. As stated above, the 3 and 4 pump RPS setpoints are not changed for O3C12.

In addition:

1. The operating limits on rod index and axial power imbalance were developed in accordance with the LOCA linear heat rate limits discussed in Chapter 7. These operational limits are provided in the Oconee 3 Cycle 12 Core Operational Limits Report.

2. Due to the lower worth of the gray APSRs, operational limits for the Group 8 control rods are not required for Cycle 12. Administrative position and maneuvering limits for the APSRs, however, will be utilized to ensure adequate fuel performance during Cycle 12 operation.

Based on the Technical Specifications derived from the analyses presented in this report, the Final Acceptance Criteria ECCS limits will not be exceeded, nor will the thermal design criteria be violated.

Figure 8-1

Core Protection Safety Limits For Oconee 3 Cycle 12

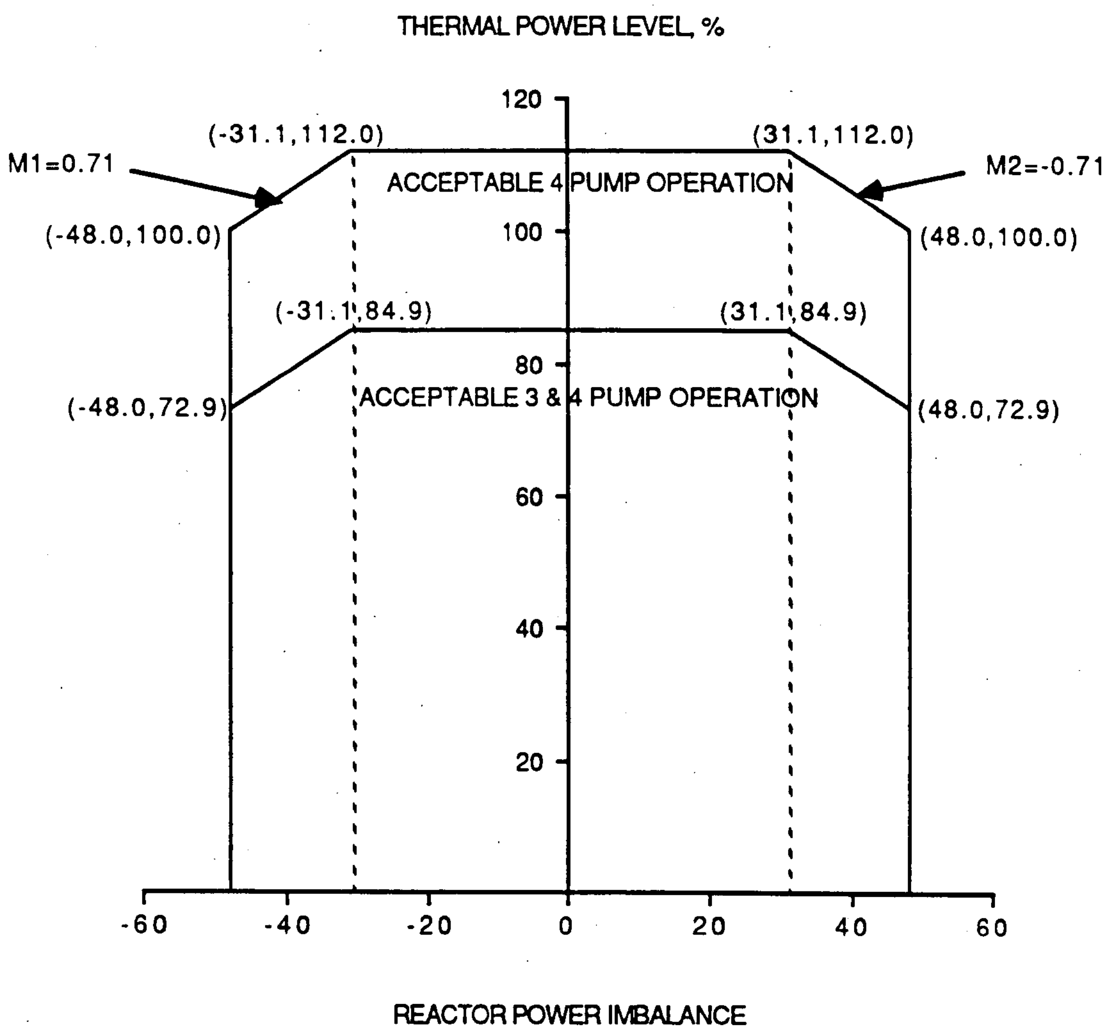
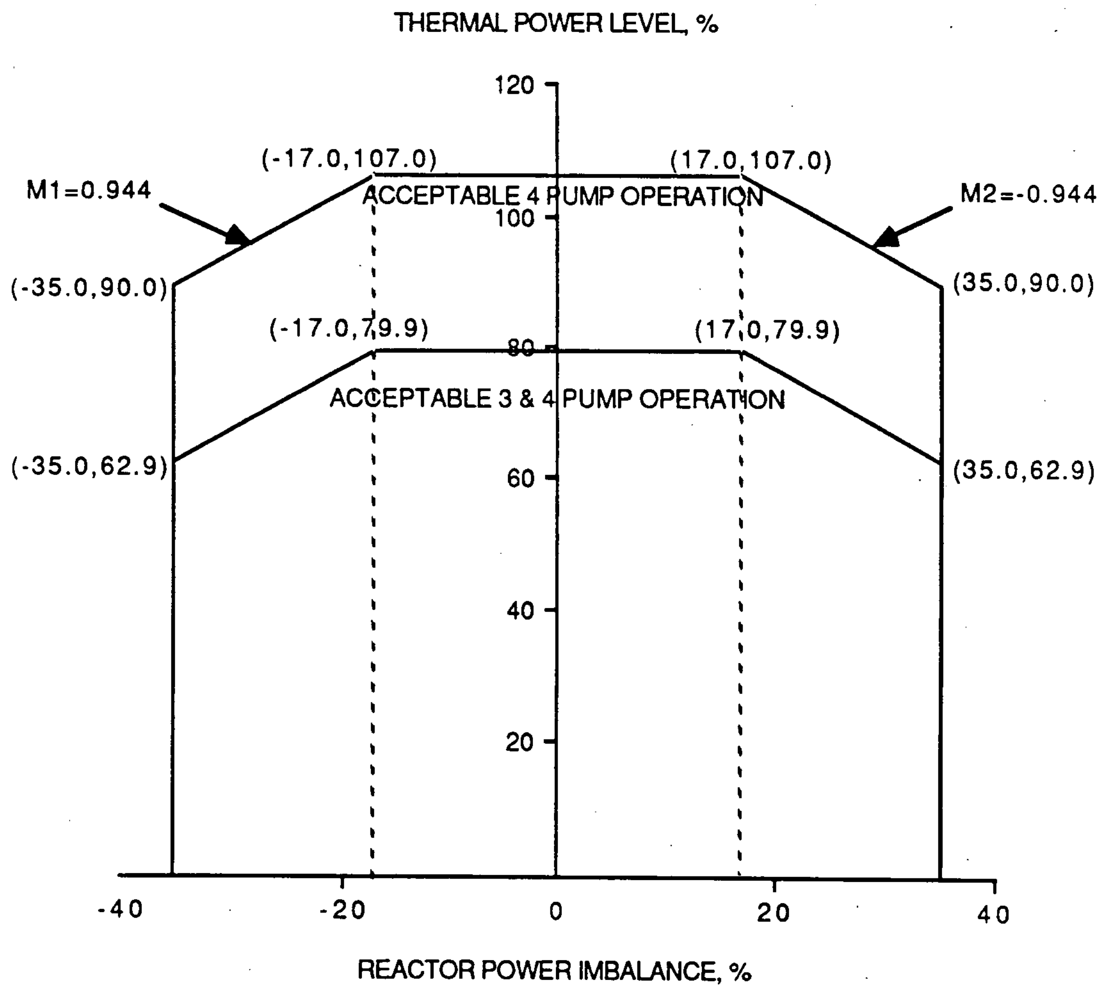


Figure 8-2

Protective System Maximum Allowable Setpoints For Oconee 3 Cycle 12



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