OCONEE UNIT 3 CYCLE 11

- Reload Report -

DPC - RD - 2011

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the eleventh 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 11 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 10 and 11 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 11 operation. In those cases where Cycle 11 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 required for Cycle 11 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 11 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 11, is the currently operating Cycle 10. Cycle 10 achieved initial criticality on March 29, 1987 and power escalation commenced on March 30, 1987. The fuel cycle design length for Cycle 11 - 410 EFPD - is based on a Cycle 10 length of 440 EFPD.

Cycle 11 will operate in a feed-and-bleed mode for its entire design length, as did Cycle 10.

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 11 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 11. Forty-three of the Batch 11 assemblies will be discharged at the end of Cycle 10 along with Batch 10B. The remaining 25 Batch 11 assemblies, designated "11B," and the fresh Batch 13 FAs - with initial enrichments of 3.22 and 3.48 wt % ^{235}U, respectively - will be loaded into the central portion of the core. Batches 8C, 9C, and 12 - with initial enrichments of 3.07, 3.18, and 3.22 wt % ^{235}U, will occupy primarily the core periphery. Figure 3-2 is a quarter-core map showing the assembly burnup and enrichment distribution at the beginning of Cycle 11.

Cycle 11 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 60 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 axial power distribution. The Cycle 11 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The Cycle 11 locations and enrichments of the BPRAs are shown in Figure 3-4.

FIGURE 3.1. CORE LOADING DIAGRAM FOR OCONEE 3, CYCLE 11

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						#011 9C	F11 12	G08 12	F05 12	#005 9C					
B				#K06 9C	E12 12	010 12	13	E08 11B	13	006 12	E04 12	#K10 9C			
C			$N12 8C	13	K02 12	13	K06 12	13	K10 12	13	K14 12	13	$N04 8C		
D		#F09 9C	13	D07 12	13	P11 11B	13	P06 11B	13	P05 11B	13	G12 12	13	#F07 9C	
E		N05 12	B09 12	13	$P08 8C	13	$K03 8C	C04 12	$K13 8C	13	$H02 8C	13	B07 12	N11 12	
F	#M13 9C	L13 12	13	M14 11B	13	D02 11B	13	M13 11B	13	B12 11B	13	M02 11B	13	L03 12	#M03 9C
G	M06 12	13	F09 12	13	$C09 8C	13	G04 12	C12 12	D09 12	13	$C07 8C	13	F07 12	13	M10 12
H	H07 12	H05 11B	13	L14 11B	N03 12	C11 11B	D03 12	R08 11B	N13 12	005 11B	D13 12	F02 11B	13	H11 11B	H09 12
K	E06 12	13	L09 12	13	$009 8C	13	N07 12	004 12	K12 12	13	$007 8C	13	L07 12	13	E10 12
L	#E13 9C	F13 12	13	E14 11B	13	P04 11B	13	E03 11B	13	N14 11B	13	E02 11B	13	F03 12	#E03 9C
M		D05 12	P09 12	13	$H14 8C	13	$G03 8C	012 12	$G13 8C	13	$B08 8C	13	P07 12	D11 12	
N		#L09 9C	13	K04 12	13	B11 11B	13	B10 11B	13	B05 11B	13	N09 12	13	#L07 9C	
O			$D12 8C	13	G02 12	13	G06 12	13	G10 12	13	G14 12	13	$D04 8C		
P				#G06 9C	M12 12	C10 12	13	M08 11B	13	C06 12	M04 12	#G10 9C			
R						#C11 9C	L11 12	K08 12	L05 12	#C05 9C					

Y

Z

YXX X	

XX = Previous cycle location

X = Batch No.

Y = $: Previous location in 03C7

= #: Previous location in 03C8

= Blank: Previous location in 03C10

FIGURE 3.2 ENRICHMENT & BURNUP FOR OCONEE 3, CYCLE 11

	8	9	10	11	12	13	14	15
H	3.22 21761	3.22 13853	3.22 26535	3.22 13854	3.22 22291	3.48 0	3.22 29354	3.22 17667
K	3.22 13853	3.22 17188	3.48 0	3.07 30104	3.48 0	3.22 17905	3.48 0	3.22 17758
L	3.22 26535	3.48 0	3.22 19867	3.48 0	3.22 21575	3.48 0	3.22 16964	3.18 30313
M	3.22 13854	3.07 30105	3.48 0	3.07 27286	3.48 0	3.22 15192	3.22 17452	
N	3.22 22291	3.48 0	3.22 21580	3.48 0	3.22 17205	3.48 0	3.18 30576	
O	3.48 0	3.22 17903	3.48 0	3.22 15198	3.48 0	3.07 29025		
P	3.22 29355	3.48 0	3.22 16982	3.22 17505	3.18 30599			
R	3.22 17662	3.22 17805	3.18 30285					

S_L

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

FIGURE 3.3 CONTROL ROD LOCATIONS FOR OCONEE 3, CYCLE 11

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B						4		6		4					
C					2		5		5		2				
D				7		8		7		8		7			
E			2		5		1		1		5		2		
F		4		8		6		3		6		8		4	
G			5		1		3		3		1		5		
HW —		6		7		3		4		3		7		6	— Y
K			5		1		3		3		1		5		
L		4		8		6		3		6		8		4	
M			2		5		1		1		5		2		
N				7		8		7		8		7			
O					2		5		5		2				
P						4		6		4					
R															

Z

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 11

	8	9	10	11	12	13	14	15
H						1.1		
K			1.1		1.1		0.0	
L		1.1		1.1		0.8		
M			1.1		1.1			
N		1.1		1.1		0.0		
O	1.1		0.8		0.0			
P		0.0						
R								

X.X	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 11, 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 B5Z fuel assemblies.

The Batch 13 MK-B7 fuel assemblies are similar in design to the MK-B6 fuel first demonstrated in the Oconee 1 Cycle 11 Reload (Reference 2). Like the MK-B6 fuel design, the MK-B7 features intermediate Zircaloy grids, a skirtless inconnel upper end grid (dimensionally equivalent to the intermediate Zircaloy grids), and a removable upper end fitting. New features for the MK-B7 include slightly longer fuel rods and a shorter lower end fitting. The longer fuel rods have increased plenum volume allowing for higher fuel burnup. At the same time, a net increase in shoulder gap (fuel rod to nozzle gap) has also been included to provide additional margin for fuel rod growth.

The Oconee 3 Cycle 11 core will have 60 BPRAs inserted in the Batch 13 fuel assemblies. Forty-four (44) of the BPRAs will be new, and the remaining 16 will be reinserted from the Cycle 10 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 11 design. All methods are consistent with the approved methodologies of Reference 3 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 Batch 9C is more limiting than other batches due to its longer previous incore exposure time. The Batch 9C assembly power histories were assessed against Duke's generic creep collapse analysis which is based on the CROV computer code and procedures described in topical report BAW-10084, Rev. 2[4]. The TACO2[5] code was used to calculate internal pin pressure and clad temperatures used as input to CROV. The collapse time for the most limiting assembly was conservatively determined to be 31,400 EFPH, which is greater than the maximum projected residence time of Cycle 11 fuel (Table 4-1).

4.2.2 Cladding Stress

As described in Reference 3, 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 3 except that the static stress analysis uses design stress intensity limits on mechanical properties based on the requirements of ASME Code Article III-2000. 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 regards to stress

4.2.3 Cladding Strain

Duke has performed a cladding strain calculation using TACO2[5] in accordance with the approved methodology[3]. This analysis demonstrated that the uniform, circumferential strain of the cladding was within 1.0%.

4.3 Thermal Design

All fuel in the Cycle 11 core is thermally similar. The fresh Batch 13 fuel inserted for Cycle 11 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 determined separately for each batch of fuel using the TACO2[5] computer code. The fuel parameters used to determine the fuel melt limits are shown in Table 4-2.

The input shown 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 42,584 MWd/mtU and the maximum fuel rod burnup is predicted to be 43,990 MWd/mtU. Fuel rod internal pressure has been evaluated using TACO2[5] 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 13 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 13 fuel assemblies is identical to those of the present fuel.

Table 4-1.

Fuel Design Parameters and Dimensions

	Batch No.				
	8C	9C	11B	12	13
FA type	Mark B4	Mark B5	Mark B5Z	Mark B5Z	Mark B7
No. of FAs	16	16	25	60	60
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	3.07	3.18	3.22	3.22	3.48
Est. residence time, EOC 11, EFPH	29,240	30,378	29,249	20,880	10,080
Cladding collapse time, EFPH	>31,400	>31,400	>34,000	>29,800	>15,000

Table 4-2. Linear Heat Rate to Melt Analysis

	Batch No.				
	8C	9C	11B	12	13
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 het rate capability from 0-1000 MWD/MTU, kW/ft (b)	20.15	20.15	20.15	20.15	20.15
Linear heat rate capability >1000 MWD/MTU, kW/ft (b)	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, 96.5% TD @ 4000 MWD/MTU, nominal pellet and cladding dimensions.

(b) These values are utilized as fuel design limits for Cycle 11.

5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 compares the core physics parameters of design Cycles 10 and 11; the values for Cycles 10 and 11 were generated by Duke Power Company using methods described in Reference 3. 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 the eleventh cycle at full power with equilibrium xenon and normal rod positions.

The primary reasons for the differences in the physics parameters between Cycles 10 and 11 are the different shuffle patterns and the 32 Mark B assemblies inserted in Cycle 11 from the spent fuel pool. Differences in ejected and stuck rod worths between cycles are due to changes in the radial flux and burnup distributions. Calculated ejected rod worths and their adherence to criteria are considered at all times in life and at all power levels in the development of the rod position limits presented in the Technical Specifications. All safety criteria associated with these rod worths are met. The adequacy of the shutdown margin with Cycle 11 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 10 Reload Report.[6]

5.2 Analytical Input

The Cycle 11 incore measurement calculation constants to be used to compute core power distributions were obtained in a similar manner for Cycle 11 as for the reference cycle.

5.3 Changes in Nuclear Design

The methodology described in Reference 3 has been implemented for Oconee 3 Cycle 11.

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

	Cycle 10[b]	Cycle 11[c]
Cycle length, EFPD	440	410
Cycle burnup, MWd/mtU	13,763	12,831
Average core burnup, EOC, MWd/mtU	24,631	27,169
Initial core loading, mtU	82.1	82.1
Critical boron - BOC (no xenon), ppm		
HZP, Groups 7 and 8 at nominal positions[d]	1752	1635
HFP, Groups 7 and 8 at nominal positions	1516	1408
Critical boron - EOC (equilibrium xenon), ppm		
HZP, Groups 7 and 8 at nominal positions	430	391
HFP, Groups 7 and 8 at nominal positions	57	5
Control rod worths - HFP, BOC, % Δk/k		
Group 7	0.96	0.99
Group 8 (35-100% WD)	0.16	0.16
Control rod worths - HFP, EOC, % Δk/k		
Group 7 (e)	1.08	1.06
Group 8 (35-100% WD)	0.22	(f)
Max ejected rod worth - HZP, % Δk/k		
BOC, groups 5-8 inserted (g)	0.40(N12)	0.36(L10)
EOC, groups 5-8 inserted (h)	0.48(N12)	0.41(L10)
Max stuck rod worth - HZP, % Δk/k		
BOC (N12) groups 1-7 inserted (g)	1.55	1.37
EOC (N12) groups 1-7 inserted (h)	2.04	1.54
Power deficit, HFP to HZP, % Δk/k		
BOC	1.70	1.87
EOC (e)	3.16	3.21
Doppler coeff - HFP, 10^{-5} (Δk/k-°F)		
BOC (equilibrium xenon)	-1.35	-1.20
EOC (equilibrium xenon) (e)	-1.73	-1.54

Table 5-1. (Cont'd)

	Cycle 10(b)	Cycle 11(c)
Moderator coeff - HFP, 10^{-4} ($\Delta k/k$-°F)		
BOC (NOXE)	-0.47	-0.93
EOC (equilibrium xenon) (e)	-3.30	-3.40
Boron worth - HFP, ppm/% $\Delta k/k$		
BOC	124	127
EOC (e)	110	115
Xenon worth - HFP, % $\Delta k/k$		
BOC (4 days)	2.43	2.59
EOC (equilibrium) (e)	2.58	2.74
Effective delayed neutron fraction - HFP		
BOC	0.00617	0.00609
EOC	0.00518	0.00515

(a) Cycle 11 data are for the conditions stated in this report. The Cycle 10 core conditions are identified in Reference 6.

(b) Based on a 349-EFPD Cycle 9. (Actual Cycle 9 length 348.72 EFPD)

(c) Based on a Cycle 10 length of 440-EFPD.

(d) Nominal positions are as follows:

	Cycle 10	Cycle 11
HZP (BOC)	Group 7 at 100% WD, 8 at 25.0% WD	Group 7 at 100% WD, 8 at 35.0% WD
HFP (BOC)	Group 7 at 92% WD, 8 at 35% WD	Group 7 at 92% WD, 8 at 35% WD
HZP (EOC)	Group 7 at 100% WD, 8 at 25.0% WD	Group 7 at 100% WD, 8 at 100.0% WD
HFP (EOC)	Group 7 at 100% WD, 8 at 35% WD	Group 7 at 92% WD, 8 at 100% WD

(e) Group 8 = 35% WD for Cycle 10 and 100% WD for Cycle 11.
(f) Group 8 = 100% WD; therefore, there is no rod worth at EOC.
(g) Group 8 = 25% WD for Cycle 10 and 35% WD for Cycle 11.
(h) Group 8 = 25% WD for Cycle 10 and 100% WD for Cycle 11.

Table 5-2. Shutdown Margin Calculation for Oconee 3, Cycle 11

	BOC, % Δk/k	EOC % Δk/k
Available Rod Worth		
Total rod worth, HZP	7.74	8.30
Worth reduction due to poison burnup	-0.42	-0.42
Maximum stuck rod, HZP	-1.37	-1.54
Net worth	5.95	6.34
Less 10% uncertainty	-0.60	-0.63
Total available worth	5.35	5.71
Required Rod Worth		
Power deficit, HFP to HZP	1.87	3.21
Max inserted rod worth, HFP	0.33	0.51
Total required worth	2.20	3.72
Shutdown Margin		
Total available worth minus total required worth	3.15	1.99

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

FIGURE 5-1
OCONEE 3, CYCLE 11
TWO DIMENSIONAL
RELATIVE POWER DISTRIBUTION

HFP, 004 EFPD, EQXE
NOMINAL ROD POSITIONS

	8	9	10	11	12	13	14	15
H	1.108	1.233	1.039	1.083	1.113	1.379	0.855	0.476
K	1.233	1.253	1.371	0.926	1.369	1.252	1.201	0.472
L	1.039	1.371	1.225	1.335	1.131	1.359	0.860	0.257
M	1.083	0.926	1.335	1.010	1.355	1.103	0.571	
N	1.113	1.369	1.131	1.355	1.155	1.040	0.288	
O	1.379	1.252	1.359	1.103	1.040	0.375		
P	0.855	1.201	0.860	0.571	0.288			
R	0.476	0.472	0.257					

6. THERMAL-HYDRAULIC DESIGN

The generic Mark-B and Mark-BZ thermal-hydraulic design analyses supporting Cycle 11 operation were performed by Duke Power Company using the methods described in References 1, 3, 6, and 7. The Cycle 10 and Cycle 11 maximum design conditions are summarized in Table 6-1.

The Cycle 11 transition core will include 60 fresh Mark-B7 Batch 13 fuel assemblies, all of which contain BPRAs. Two asemblies will contain regenerative neutron sources, leaving 46 fuel assemblies with open guide tubes. This results in a core bypass flow of 7.9% 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 11 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 presence of Mark-BZ and Mark-B assemblies in a core results in less coolant flow in the Mark-BZ fuel then would occur in an all Mark-BZ core. The generic Mark-BZ analyses conservatively account for this transition core effect.

The Mark-B7 fuel assembly has a slightly higher pressure drop than the Mark-BZ assembly as a result of the increased resistance of the Inconel upper end grid and reconstitutable upper end fitting. The shorter Mark-B7 lower end fitting has a negligible impact on the assembly pressure drop. The Cycle 11 Mark-B7 transition core was analyzed at the limiting thermal design conditions and the MDNBR is greater than the BWC CHF correlation limit of 1.18, Reference 8.

In a Mark-BZ transition core the limiting Mark-B hot channel will receive more coolant 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 11 transition core.

No fuel rod bow penalty was included in the DNBR limit used in the generic Mark-BZ analyses, as justified in Reference 9. 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 11 operation. A conservative transition core two pump coastdown analysis was performed based on a 1.07 flux to flow setpoint and the reference design radial-local peaking factor, $F_{\Delta H} = 1.714$. The two pump coastdown is initiated from a power level of at least 106% full power. This initial condition is based on the combined effects of a 2% allowance for the secondary side heat balance uncertainty, a 2% allowance for calibration of the nuclear instrumentation to the secondary side heat balance, and a 2% allowance for transient nuclear instrumentation uncertainty. The minimum DNBR determined in the transition core flux to flow analysis is greater than the BWC CHF correlation limit of 1.18, Reference 8. The minimum DNBR determined in the generic Mark-B flux to flow analysis, conservatively based on a 1.08 flux to flow setpoint, is greater than the BAW-2 CHF correlation limit of 1.30.

Table 6-1. Thermal Hydraulic Design Conditions

	Cycle 10	Cycle 11
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design flow	106.5	109.5
Core bypass flow, % total flow[a]	7.9	7.9
Vessel inlet/outlet coolant temp at 100% power, °F	555.6/602.4	556.2/601.8
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/h-ft^2	176[b]	176[b]
CHF correlation	BAW-2/BWC	BAW-2/BWC
Min DNBR with densification penalty	2.05/>1.74	>2.05/>1.74
Hot channel factors: Enthalpy rise	1.011/1.011	1.011/1.011
Heat flux	1.014/1.014	1.014/1.014
Flow area	0.98/0.97	0.98/0.97

[a] Generic analyses based on $\geq$8.0% core bypass flow.

[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 10 parameters to determine the effect of the Cycle 11 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 7. Since Batch 13 reload fuel assemblies contain fuel rods with a theoretical density higher than those considered in Reference 7, the conclusions in that reference are still valid.

No new dose calculations were performed for the reload report. The dose considerations in Reference 14 are characteristic for Oconee 3 Cycle 11 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 11 are given in Table 4-2. Table 6-1 compares the Cycle 10 and 11 thermal-hydraulic maximum design conditions. Table 7-1 compares the key kinetics parameters from the FSAR and Cycle 11. 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.[15] 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. This study is reported in BAW-10103, Rev. 3.[12] 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.[10]

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[10] is conservative compared to those calculated for this reload. In addition, it has been determined that the slightly lower prepressurization of the Batch 13 fuel rods does not have an adverse impact on the LOCA analyses.[11] Thus, the analysis and the LOCA limits reported in BAW-1915 provide conservative results for the operation of Oconee 3, Cycle 11 fuel.[13]

The LOCA kW/ft limits have been reduced for the first 25 EFPDs. The LOCA 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 11 fuel after 25 EFPD.

From the examination of Cycle 11 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 11. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of Cycle 11 is considered to be bounded by previously accepted analyses. The intitial conditions of the transients in Cycle 11 are bounded by the FSAR and/or the fuel densification report.[7]

Table 7.1 Comparison of Key Parameters for Accident Analysis

Parameter	FSAR[1] value	Predicted Cycle 11 value
BOC Doppler coeff, 10^{-5}, $\Delta k/k/°F$	-1.17	-1.20
EOC Doppler coeff, 10^{-5}, $\Delta k/k/°F$	-1.33[a]	-1.54
BOC moderator coeff, 10^{-4}, $\Delta k/k°F$	+0.5[b]	-0.93
EOC moderator coeff, HZP, 10^{-4}, $\Delta k/k/°F$	-3.0[c]	-3.00
, HFP, 10^{-4}, $\Delta k/k/°F$	-3.5[c]	-3.40
All rod bank worth, HZP, % $\Delta k/k$	10.0	8.30
Boron reactivity worth, 70°F ppm/1% $\Delta k/k$	75	91
Max. ejected rod worth, HFP, % $\Delta k/k$	0.65	0.23
Dropped rod worth, HFP, % $\Delta k/k$	0.40	0.12
Initial boron conc, HFP, ppm	1400	1408[d]

(a) -1.2×10^{-5} $\Delta k/k/F$ was used for steam line break analysis.

-1.3×10^{-5} $\Delta k/k/F$ was used for cold water accident (pump start-up).

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

(c) The HZP moderator temperature coefficient is one of the key parameters assummed 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 11.

Table 7-2. LOCA Limits, Oconee 3, Cycle 11 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 11, 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 11 operation in accordance with the methods of Reference 3 to account for minor changes in power peaking and control rod worths, and beginning of cycle boron concentration requirements.

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.

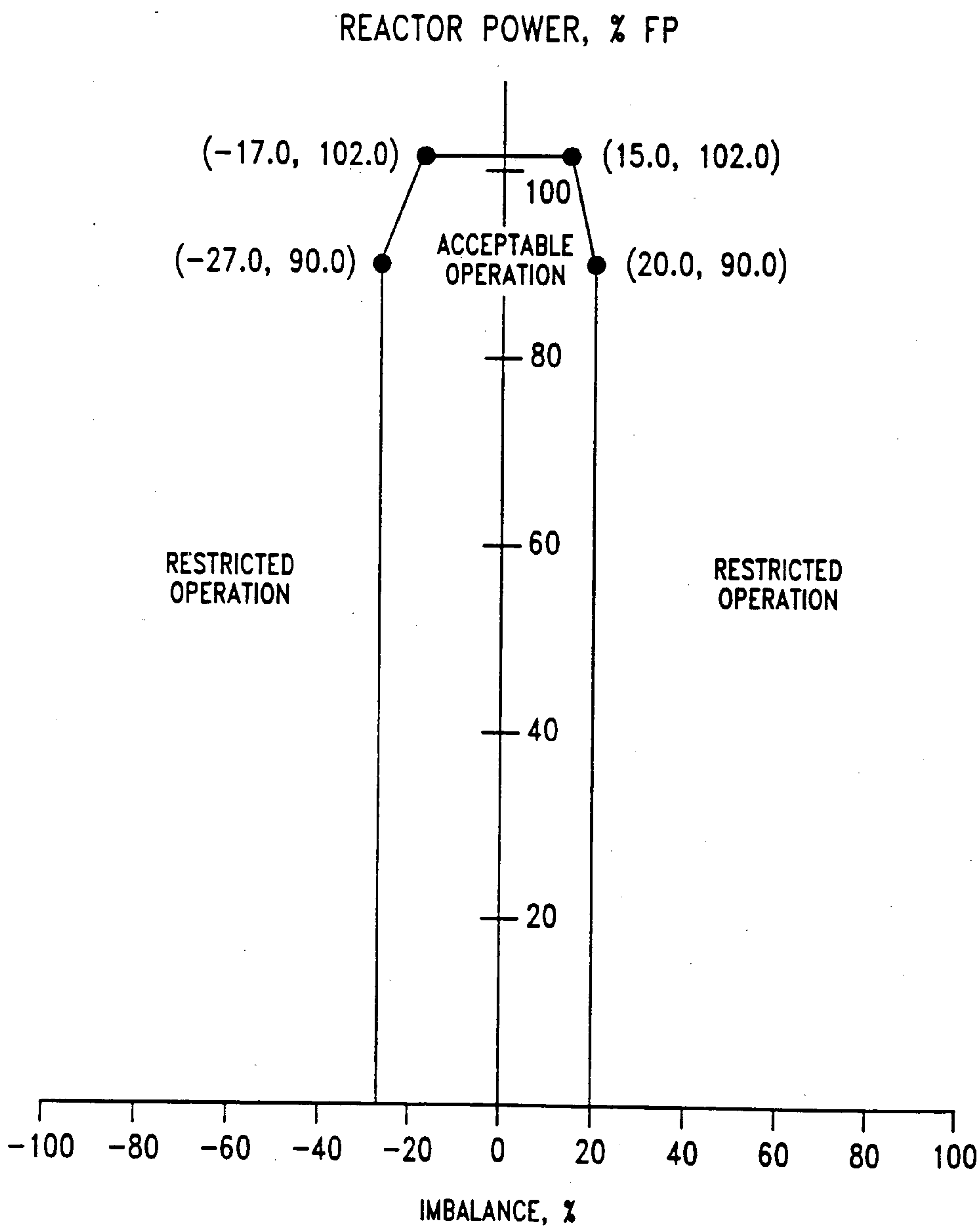
2. Due to the lower worth of the gray APSRs, operational limits for the Group 8 control rods are not required for Cycle 11. Administrative position and maneuvering limits for the APSRs, however, will be utilized to ensure adequate fuel performance during Cycle 11 operation.

3. The minimum required boron concentration in the BWST has been changed from 1835 ppm to 1950 ppm. This increase will ensure that adequate shutdown margin exists during refueling operations. In addition, the minimum required CBAST volume has been increased from 1020 ft^3 to 1100 ft^3. This increase will ensure that the CBAST can borate the Reactor Coolant System to 1% $\Delta k/k$ subcritical at cold conditions with the maximum worth stuck rod and no credit for xenon at the worst time in core life.

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 is a revision to a previous Technical Specification limit.

Figure 8-1

Operational Power-Imbalance Limits, 0 EFPD to EOC



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