Duke Power
A Duke Energy Company

W. R. McCollum, Jr.
Vice President

Duke Power
Oconee Nuclear Site
7800 Rochester Highway
Seneca, SC 29672
(864) 885-3107 OFFICE
(864) 885-3564 FAX

May 30, 2000

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-001

Attention: Document Control Desk

Subject: Oconee Nuclear Station, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
License Amendment Request (TSC 2000-003),
Supplement 1.

Reference:

1) Letter, W. R. McCollum, Jr. (Duke) to U. S. NRC Document Control Desk, "License Amendment Request (TSC 2000-03), Implementation of Mark-B11 Fuel with M5 Cladding," April 13, 2000.

In Reference 1, Duke Energy Corporation (Duke) submitted a license amendment request (LAR) to change the Technical Specifications and Bases to implement Mark-B11 fuel with M5 Cladding. As part of that LAR, associated design related information and the draft Core Operating Limits Report (COLR) for Oconee Unit 1 Cycle 20 (O1C20) were provided as background information only. Attachment VII to the LAR contained information related to the design of O1C20. That cycle is the first Oconee core scheduled to implement the new fuel design. Since the time of the original submittal, the core design has been modified to accommodate a reduction in the projected burnups for Cycle 19 and Cycle 20. As a result of the revised core design, some of the background information contained in Attachment VII has changed. The attachment to this letter simply provides the revised design information for O1C20.

A001

The most significant change is a reduction in the feed batch fuel enrichment by approximately 0.2 weight percent ^{235}U. Others modifications include a small adjustment in the core loading pattern, a slight reduction in the burnable poison rod assemblies (BPRA) concentrations, and a change in the projected assembly burnups.

The change in the core design does not alter the conclusions drawn in the original submittal. As such, this supplement does not affect the No Significant Hazards Consideration and Environmental Assessment/Impact Statement for the LAR.

Pursuant to 10 CFR 50.91, a copy of this supplement is being sent to the State of South Carolina.

Please address any comments or questions regarding this matter to Edwin D. Price Jr. at (864) 885-4388.

Very truly yours,

W. R. McCollum, Jr., Site Vice President
Oconee Nuclear Site

Attachment

xc: L. A. Reyes, Regional Administrator,
U. S. Nuclear Regulatory Commission, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

D. E. Labarge, NRC Senior Project Manager (ONS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

M. C. Shannon
Senior Resident Inspector
Oconee Nuclear Station

V. R. Autry, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
South Carolina Department of Health and
Environmental Control
2600 Bull Street
Columbia, SC 29201

AFFIDAVIT

W. R. McCollum, Jr., being duly sworn, states that he is the Site Vice President of Duke Energy Corporation: that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission revisions to the Oconee Nuclear Station Facility Operating Licenses No. DPR-38, DPR-47, and DPR-55; and that all the statements and matters set forth herein are true and correct to the best of his knowledge.

W. R. McCollum, Jr., ~~Site~~ Vice President

Subscribed and sworn to be before me this 30th day of

May, 2000.

Conice M Dreazale
Notary Public

My Commission Expires:

2/12/2003

SEAL

ATTACHMENT VII, 5/30/00 Update

DESIGN INFORMATION RELATIVE TO
OCONEE UNIT 1 CYCLE 20

TABLE OF CONTENTS

List of Tables

List of Figures

1 INTRODUCTION AND SUMMARY

This report justifies the operation of the twentieth cycle of Oconee Nuclear Station, Unit 1, at the rated core power of 2568 MWth. Included are the required analyses as outlined in the USNRC document "Guidance for Proposed License Amendments Relating to Refueling", June 1975.

Cycle 20 for Oconee Unit 1 will be the first Oconee cycle to utilize Mark-B11 fuel including M5 cladding. To support Cycle 20 operation of Oconee Unit 1, this report employs analytical techniques and design bases established in reports that have been previously submitted to the USNRC. The Duke Power non-LOCA transient analysis methods are documented in topical report DPC-NE-3005 (Reference 1).

Section 2 of this report describes the operating history for fuel in Oconee Unit 1. Section 3 is a general description of the reactor core, and the fuel system design is provided in Section 4. Reactor and system parameters and conditions are summarized in Sections 5, 6 and 7. All of the accidents analyzed in the UFSAR (Reference 2) have been reviewed for Cycle 20 operation. In those cases where Cycle 20 characteristics were conservative compared to those analyzed for the generic analysis, a new analysis was not performed. Changes to the Technical Specifications and the Core Operating Limits Report (COLR) are provided in Section 8.

The Technical Specifications have been reviewed, and the modifications for Cycle 20 are justified in this report. Based on the analyses performed, it has been concluded that Oconee Unit 1 Cycle 20 can be safely operated at a core power level of 2568 MWth.

2 OPERATING HISTORY

The reference fuel cycle for the nuclear and thermal-hydraulic analyses ofOconee Unit 1, Cycle 20, is the currently operating Cycle 19. Cycle 19 achieved initial criticality on July 6, 1999. The fuel cycle design length for Cycle 20 - 440 $\pm$ 10 EFPD - is based on an assumed Cycle 19 length of 475 $\pm$ 10 EFPD. No operating anomalies have occurred during previous cycle operations that would adversely affect fuel performance in Cycle 20.

Cycle 20 will operate in a feed-and-bleed mode for its entire design length, as did Cycle 19.

3 GENERAL DESCRIPTION

The Oconee Unit 1 reactor core and fuel design bases are described in detail in Chapter 4 of the UFSAR (Reference 2). The Cycle 20 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 Cycle 20 (Batch 22) fuel consists of dished-end, cylindrical pellets of uranium dioxide clad in M5 cladding. All other fuel assemblies consist of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The Batch 22 fuel assemblies have an average nominal fuel loading of 459.0 kg uranium. All other fuel assemblies have an average nominal fuel loading of 487.2 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 1, Cycle 20. The 12 assemblies remaining from the original 60 included in Batch 19 (3.68 wt% ^{235}U) will be designated as Batch 19C. The 45 assemblies remaining from the original 60 included in Batch 20 (3.61 wt% ^{235}U) will be designated as Batch 20B. The 60 Batch 21 (44 at 3.68 and 16 at 4.02 wt% ^{235}U, Batch 21A and 21B, respectively) assemblies will be retained along with the 60 Batch 22 feed assemblies (3.21 wt% ^{235}U). The core periphery is composed of Batch 19 and Batch 20 assemblies. The Batch 22 assemblies are distributed evenly throughout the core interior with the rest of the Batch 20 and Batch 21 assemblies. Figure 3-2 is a quarter-core map showing the fuel assembly burnup and enrichment distribution at the beginning of Cycle 20.

Cycle 20 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 20 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The Cycle 20 locations and enrichments of the BPRAs are shown in Figure 3-4.

Figure 3-1

Core Loading Diagram

Oconee 1, Cycle 20

X

W Y

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						G-15 19C	G-13 20B	H-01 19C	G-03 20B	G-01 19C					
B				N-10 20B	K-06 20B	N-07 21A	22	K-08 21A	22	N-09 21A	K-10 20B	N-06 20B			
C			N-12 20B	22	P-09 21B	22	C-10 21A	22	C-06 21A	22	P-07 21B	22	N-04 20B		
D		L-12 20B	22	O-08 21A	22	M-12 21A	22	O-11 20B	22	M-04 21A	22	H-03 21A	22	L-04 20B	
E		F-09 20B	K-14 21B	22	O-05 20B	22	L-05 21A	22	L-11 21A	22	E-03 20B	22	K-02 21B	F-07 20B	
F	R-07 19C	G-12 21A	22	N-11 21A	22	L-02 20B	D-13 21B	P-06 20B	D-03 21B	B-06 20B	22	N-05 21A	22	G-04 21A	R-09 19C
G	O-07 20B	22	L-03 21A	22	E-10 21A	O-04 21B	E-08 21A	22	H-11 21A	O-12 21B	E-06 21A	22	L-13 21A	22	O-09 20B
H	R-08 19C	H-09 21A	22	E-13 20B	22	L-14 20B	22	G-09 20B	22	F-02 20B	22	M-03 20B	22	H-07 21A	A-08 19C
K	C-07 20B	22	F-03 21A	22	M-10 21A	C-04 21B	H-05 21A	22	M-08 21A	C-12 21B	M-06 21A	22	F-13 21A	22	C-09 20B
L	A-07 19C	K-12 21A	22	D-11 21A	22	P-10 20B	N-13 21B	B-10 20B	N-03 21B	F-14 20B	22	D-05 21A	22	K-04 21A	A-09 19C
M		L-09 20B	G-14 21B	22	M-13 20B	22	F-05 21A	22	F-11 21A	22	C-11 20B	22	G-02 21B	L-07 20B	
N		F-12 20B	22	H-13 21A	22	E-12 21A	22	C-05 20B	22	E-04 21A	22	C-08 21A	22	F-04 20B	
O			D-12 20B	22	B-09 21B	22	O-10 21A	22	O-06 21A	22	B-07 21B	22	D-04 20B		
P				D-10 20B	G-06 20B	D-07 21A	22	G-08 21A	22	D-09 21A	G-10 20B	D-06 20B			
R						K-15 19C	K-13 20B	H-15 19C	K-03 20B	K-01 19C					

Z

XX X	Previous Cycle Location Batch Number

Figure 3-2

Enrichment and Burnup

Oconee 1, Cycle 20

	8	9	10	11	12	13	14	15
H	3.61 35678	3.21 0	3.61 28937	3.21 0	3.61 28411	3.21 0	3.68 19184	3.68 35470
K	3.21 0	3.68 19733	4.02 14334	3.68 20191	3.21 0	3.68 18149	3.21 0	3.61 33405
L	3.61 28937	4.02 14358	3.61 28925	3.21 0	3.68 19301	3.21 0	3.68 20002	3.68 38633
M	3.21 0	3.68 20192	3.21 0	3.61 28403	3.21 0	4.02 15920	3.61 30319	
N	3.61 28411	3.21 0	3.68 19279	3.21 0	3.68 18769	3.21 0	3.61 36065	
O	3.21 0	3.68 18150	3.21 0	4.02 15883	3.21 0	3.61 33720		
P	3.68 19230	3.21 0	3.68 20006	3.61 30347	3.61 36058			
R	3.68 35468	3.61 33381	3.68 38648					

X.XX XXXXX	Initial Enrichment in wt% ^{235}U BOC Burnup in MWd/mtU

Figure 3-3

Control Rod Locations

Oconee 1, Cycle 20

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		
W H		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			**Group Number**		
R															

Z

Group	# of Rods	Function	Group	# of Rods	Function
1	8	Safety	5	12	Control
2	8	Safety	6	8	Control
3	8	Safety	7	8	Control
4	9	Safety	8	8	APSRs
			Total:	**69**	

Figure 3-4

BPRA Enrichment and Distribution

Oconee 1, Cycle 20

	8	9	10	11	12	13	14	15
H		1.1		1.1		0.8		
K	1.1				1.1			
L				1.1		0.5		
M	1.1		1.1		0.8			
N		1.1		0.8				
O	0.8		0.5					
P								
R								

X.X	BPRA Concentration in wt% B_4C in Al_2O_3

4 FUEL SYSTEM DESIGN

4.1 Fuel Assembly Mechanical Design

The reinsert fuel is comprised of Mark-B10F and Mark-B10L fuel. The Mark-B10F (previously Mark-B10T) design was presented in the Oconee 3 Cycle 16 reload report (Ref. 3). The Mark-B10L has radial zoned enrichment and a quick disconnect upper end fitting, but is otherwise similar to the Mark-B10F design.

Oconee 1 Cycle 20 will contain the first reload of Mark-B11 fuel. The Mark-B11 design offers improvements in departure from nucleate boiling (DNB) margins and fuel cycle economy while retaining many proven features of the earlier fuel assembly designs. These features include: keyable spacer grids, floating grid restraint system, flow-optimized control rod guide tube assembly, quick disconnect upper end fitting, anti-straddle lower end fitting, Zircaloy intermediate grids, cruciform holddown spring, and debris resistant fuel rods (extended lower end plug on fuel rods).

The primary design changes, which enhance nuclear, thermal-hydraulic and mechanical performance, include the following:

1. Reduced diameter fuel rod,
2. Flow mixing vanes on five of the six intermediate spacer grids,
3. Improved grid restraint system, and
4. M5 fuel rod cladding

The reduced fuel pin diameter increases uranium utilization, which improves fuel cycle economy. Mixing vane grids increase DNB margin by improving the flow mixing. Grid restraint improvements provide additional structural strength to accommodate the increased hydraulic loads from the flow mixing grids. The M5 fuel rod cladding provides additional corrosion margin.

Table 4-1 depicts fuel design parameters for the fuel operating in Oconee 1 Cycle 20.

4.2 Fuel Rod Design

The mechanical evaluation for the Mark-B10 and Mark-B11 fuel rod designs is discussed in this section.

4.2.1 Cladding Collapse

The creep collapse analysis determines the fuel rod burnup at which the cladding collapses. Therefore, the fuel rod burnup is limited to a value that does not exceed the creep collapse criteria. The methods described in BAW-10084P-A, Rev. 3 (Refs. 4 and, 5) are used to analyze cladding creep collapse.

4.2.2 Cladding Stress

Cladding stress is analyzed with the methods described in BAW-10186P-A (Refs. 6, 7, and 8) and BAW-10179-A (Ref. 9). The analyses show the cladding stresses to be within the limit.

4.2.3 Cladding Strain

The uniform, circumferential strain of the cladding is limited to 1.0% (Ref. 8). The methods described in DPC-NE-2008P-A (Ref. 10) are used to analyze cladding strain. This analysis determines conservative limits on linear heat rate, which ensures that the cladding strain will be less than 1.0%.

4.2.4 Cladding Fatigue

The cladding is limited to a cumulative fatigue usage factor of 90 percent (Ref. 8). Cladding fatigue is analyzed with the methods described in BAW-10186P-A (see Ref. 6). The analyses show that the cumulative usage factor is below the limit.

4.3 Thermal Design

Conservative limits on linear heat rate are used to prevent the centerline fuel temperature from exceeding the fuel melting point. The methods described in DPC-NE-2008P-A are used to determine the linear heat rates that result in centerline fuel melt. Representative limits on linear heat rate are depicted in Table 4-1.

The methods described in DPC-NE-2008P-A are also used to analyze internal fuel rod pressure. The fuel rod pressure is limited to a proprietary value over nominal system pressure or must be less than the pressure that causes cladding liftoff (whichever is more conservative). This analysis determines the fuel rod burnup at which these criteria are exceeded. Therefore, the fuel rod burnup is limited to a value that does not exceed the internal fuel rod pressure criteria.

4.4 Cladding Corrosion

Per Reference 6, cladding corrosion is analyzed with the methods and oxide limit defined in Reference 11. The analyses show that fuel cladding oxide is below the 100 micron limit at the end of Cycle 20.

4.5 Material Compatibility

The Mark-B10F/B10L fuel assemblies and the structural components of the Mark-B11 assemblies do not utilize component materials different from previous cycles. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions is identical to that of previous fuel. Reference 12 determined that there were no material compatibility issues for M5 cladding.

Table 4-1

Fuel Design Parameters and Dimensions

Batch number	19	20	21A	21B	22
Fuel assembly type	B10F	B10L	B10L	B10L	B11
Number of assemblies	12	45	44	16	60
Fuel rod OD, inches	0.430	0.430	0.430	0.430	0.416
Fuel rod ID, inches	0.380	0.380	0.380	0.380	0.368
Flex spacers, type	Spring	Spring	Spring	Spring	Spring
Rigid spacers, type	None	None	None	None	None
Undensified active fuel length, inches	142.29	142.29	142.29	142.29	143.05
Fuel pellet OD (mean spec), inches	0.3735	0.3735	0.3735	0.3735	0.3615
Fuel pellet initial density (mean spec), %TD	96.0	96.0	96.0	96.0	96.0
Initial fuel enrichment, w/o ^{235}U	3.68	3.61	3.68	4.02	3.21
Enrichment of radial zoned rods, w/o ^{235}U	-----	3.31	3.38	3.72	2.91
Axial blanket initial enrichment, w/o ^{235}U	2.00	2.00	2.00	2.00	2.00
Max. EOC pin burnup (MWd/mtU)	44,793	51,544	39,271	35,673	20,670
Average linear heat rate @ 100% of 2568 MW, kw/ft	5.72	5.72	5.72	5.72	5.69
Representative linear heat rate to melt values, kw/ft	21.2	21.2	21.2	21.2	22.0

5 NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 compares the core physics parameters of design Cycle 20 with those of the reference Cycle 19. The Cycle 19 and 20 values were generated by Duke Power Company using the CASMO-3/SIMULATE-3 based reload design methods described in Reference 13. 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 20 at full power with equilibrium xenon and nominal rod positions.

The primary reasons for the differences in the physics parameters between Cycles 19 and 20 are the variation in the shuffle pattern, fresh fuel enrichment, and previous end of cycle fuel assembly burnups for Cycle 20. 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 worth's are met. The adequacy of the shutdown margin with Cycle 20 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 incorporated since the shutdown analysis was calculated using a three-dimensional model.

5.2 Analytical Input

The Cycle 20 incore measurement calculation constants to be used to compute core power distributions were obtained using CASMO-3/SIMULATE-3 using the same process that was used for the reference cycle.

5.3 Changes in Nuclear Design

The methodology described in Reference 13 has been implemented for both Oconee 1 Cycle 20 and the reference cycle.

Table 5-1

Oconee 1 Physics Parameters [a]

	Cycle 19 [b]	Cycle 20 [c]
Cycle Length, EFPD (Nominal)	490	440
Cycle Burnup, MWd/mtU (Nominal)	14,649	13,355
Average Core Burnup, EOC, MWd/mtU (Nominal)	31,178	30,483
Initial Core Loading, mtU	85.9	84.6
Critical Boron - BOC (no xenon), ppm		
HZP, groups 7 and 8 at nominal positions	2038	1776
HFP, groups 7 and 8 at nominal positions	1855	1610
Critical Boron - EOC (equilibrium xenon), ppm		
HZP, groups 7 and 8 at nominal positions	343	297
HFP, groups 7 and 8 at nominal positions	7	8
Control Rod Worth - HFP, BOC, %Δk/k		
Group 7	0.890	0.936
Group 8[d]	0.151	0.164
Control Rod Worth - HFP, EOC, %Δk/k		
Group 7	1.050	1.069
Group 8	(e)	(e)
Max Ejected Rod Worth - HZP, %Δk/k [f]		
BOC, groups 5-8 inserted	0.373 (L10)	0.287 (L10)
EOC, groups 5-8 inserted	0.340 (L10)	0.298 (L10)
Max Stuck Rod Worth - HZP, %Δk/k		
BOC	0.961 (N12)	0.938 (N12)
EOC	1.274 (N12)	1.262 (N12)
Power Deficit, HFP to HZP, %Δk/k		
BOC	1.082	1.194
EOC	2.555	2.479
Doppler Coeff - HFP, 10^{-5} (Δk/k-°F)		
BOC (no xenon)	-1.41	-1.39
EOC (equilibrium xenon)	-1.63	-1.58

Table 5-1 (cont'd)

	Cycle 19 [b]	Cycle 20 [c]
Moderator Coeff - HFP, 10^{-4} (Δk/k-°F)		
BOC (no xenon)	-0.30	-0.33
EOC (equilibrium xenon)	-3.32	-3.20
Boron Worth - HFP, ppm/%Δk/k		
BOC	151	135
EOC	121	110
Xenon Worth - HFP, %Δk/k		
BOC (4 days)	2.52	2.58
EOC (equilibrium)	2.77	2.82
Effective delayed neutron fraction - HFP		
BOC	0.00618	0.00614
EOC	0.00514	0.00512

(a) EOC Physics Parameters are provided at the end of the burnup window (nominal + 10 EFPD) except where indicated to be nominal.
(b) Based on a 432 ± 10 EFPD Cycle 18 (Actual Cycle 18 length was 435.43 EFPD).
(c) Based on an assumed Cycle 19 length of 475 ± 10 EFPD.
(d) Worth is calculated from 35% to 100% WD for both cycles.
(e) CRGP8 = 100% WD. Therefore, there is no CRGP8 worth at EOC.
(f) Ejected rod worths for both cycles include a 15% uncertainty penalty.

Table 5-2 Shutdown Margin Calculation for Oconee 1, Cycle 20

	BOC, %Δk/k	EOC, %Δk/k
Available Rod Worth		
Total rod worth, HZP	7.841	8.390
Worth reduction due to poison burnup	-0.400	-0.400
Maximum stuck rod, HZP	-0.938	-1.262
Net worth	6.503	6.728
Less 10% uncertainty	-0.650	-0.673
Total available worth	5.853	6.055
Required Rod Worth		
Power deficit, HFP to HZP	1.194	2.479
Max inserted rod worth, HFP	0.307	0.513
SDM Boron Bias, HFP to HZP	0.308	0.505
Total required worth	1.809	3.497
Shutdown Margin		
Total available worth minus total required worth	4.044	2.558

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

Figure 5-1

Two Dimensional Relative Power Distribution

Oconee 1, Cycle 20

HFP, 004 EFPD, EQXE
NOMINAL ROD POSITIONS

	8	9	10	11	12	13	14	15
H	1.06	1.32	1.17	1.35	1.16	1.31	0.97	0.33
K	1.32	1.31	1.40	1.34	1.36	1.29	1.09	0.33
L	1.17	1.40	1.14	1.33	1.28	1.25	0.75	0.21
M	1.35	1.34	1.32	1.13	1.30	1.11	0.44	
N	1.16	1.35	1.27	1.30	1.14	0.93	0.27	
O	1.31	1.29	1.25	1.11	0.93	0.36		
P	0.97	1.08	0.75	0.44	0.27			
R	0.33	0.33	0.21					

6 THERMAL-HYDRAULIC DESIGN

The generic and cycle specific analyses supporting Oconee 1 Cycle 20 operation were performed by Duke Power Company using the methodology described in References 2, 14, 15, and 16. Oconee 1 Cycle 20 was analyzed using Duke's Statistical Core Design (SCD) methodology (Reference 16). Uncertainties on parameters that affect DNB performance are statistically combined to determine a statistical DNBR limit (SDL).

Previous Mark-B10 design fuel assemblies consisted of 0.430 inch diameter fuel rods with 2 Inconel and 6 intermediate non-mixing vane Zircaloy grids. The Mark-B11 fuel assembly design is composed of fuel pins with a 0.416 inch outside diameter, 2 Inconel grids, and 6 intermediate Zircaloy grids of which the upper 5 have mixing vanes. The higher pressure drop and higher cladding surface heat flux of the Mark-B11 design is offset by the larger flow area and the presence of mixing vane grids to result in improved thermal performance.

An SDL of 1.43 was calculated using a set of generic uncertainties specifically calculated for Mark-B10 fuel at Oconee with the BWC CHF correlation, Reference 15. Similarly, an SDL of 1.33 was calculated using a set of generic uncertainties specifically calculated for Mark-B11 fuel at Oconee with the BWU-Z CHF correlation with performance factor, Reference 17. To provide design flexibility, margin is added to the SDL to determine a design DNBR limit (DDL). The system parameter uncertainties used in DPC-NE-2005P-A, Rev. 2 (Reference 16) and given in Table 6-1 bound the uncertainties specifically calculated for Oconee. The Oconee 1 Cycle 20 nominal thermal-hydraulic design conditions are given in Table 6-2.

Oconee 1 Cycle 20 will contain the first full batch of Mark-B11 fuel assemblies at Oconee. Therefore, there are technical specification changes required for the use of Mark-B11 fuel. Oconee Technical Specifications Sections 2.1 and Bases Sections B 2.1.1 and B 3.4.1 were updated to add the BWU CHF correlation and its associated limits. Section B 2.1.1 was updated to add the topical BAW-10199P to the reference section. Section 5.6.5 was updated to reference revision 2 of DPC-NE-2005P-A. Section 1.1 of the Core Operating Limits Report (COLR) has been updated, as attached, to include revision 2 of the DPC-NE-2005P-A.

The M5 cladding alloy has no significant impact on DNB analyses and is discussed in Section 4. Technical Specifications Section 4.2.1 was updated to add M5 alloy to the discussion of the alloys used in the fuel rod cladding.

Table 6-1

System Uncertainties Included in the
Statistical Core Design Analysis

Reference 16

Parameter		Uncertainty	Distribution
Core power		+/- 2 %	Normal
RCS flow	4 Pump:	+/- 2.0 %	Normal
	3 Pump:	+/- 3.2 %	
	2 Pump:	+/- 4.2 %	
Pressure		+/- 30 psi	Normal
Inlet Temperature		+/- 2 °F	Normal

Table 6-2

Nominal Thermal-Hydraulic Design Conditions
Oconee 1 Cycle 20

		Generic	Cycle 20
Design Power level, MW_{th}		2568	2568
System pressure, psia		2200	2200
Reactor coolant flow, % design flow		107.5	107.5
Design flow, gpm		352,000	352,000
Core bypass flow, %		7.0	6.3
Vessel temperature at 100% FP, °F			
Inlet		555.8	555.8
Outlet		602.2	602.2
Reference design FΔH		1.714	1.714
Reference design axial shape		1.5 Cosine	1.5 Cosine
Active Fuel Length, inches		142.29 (B10F/L) 143.05 (B11)	142.29 (B10F/L) 143.05 (B11)
Average heat flux at 100%FP, 10^3 Btu/hr-ft^2		178 (B10F/L) 183 (B11)	178 (B10F/L) 183 (B11)
CHF correlation		BWC (B10F/L) BWU-Z, w/PF (B11)	BWC (B10F/L) BWU-Z, w/PF (B11)
Statistical DNBR limit		1.43 (B10F/L) 1.33 (B11)	1.43 (B10F/L) 1.33 (B11)
Hot Channel Factors	Power Factor, Fq	Note 1	Note 1
	Flow Area	Note 1	Note 1

Note:

1. A Fq Hot Channel Factor of 1.0132 (Mark-B10 fuel)/1.0133 (Mark-B11 fuel) and a flow area reduction of 3% was used in the derivation of the SDL.

7 ACCIDENT ANALYSIS

7.1 Safety Analysis

On February 17, 2000, all three Oconee units implemented accident analysis utilizing the Duke Power transient analysis methods. The thermal-hydraulic system transients, for these methods, are based upon those provided in Reference 1. For Oconee 1 Cycle 20, each of the UFSAR accident analysis has been evaluated for the change in the feed batch fuel design. Oconee 1 Cycle 20 will be the first core to utilize a full batch of Mark-B11 fuel. The smaller fuel rod diameter of the Mark-B11 design along with the mixing vane grids results in flow diversion during the transition cores. Transition core penalties have been developed and are applied to the mixed cores for both the Mark-B10 and the Mark-B11 DNB limits.

Each UFSAR accident listed has been evaluated with respect to the changes in feed batch fuel design and the Oconee 1 Cycle 20 reload parameters.

- Startup Accident
- Rod Withdrawal At Power Accident
- Moderator Dilution Accident
- Cold Water Accident
- Loss of Coolant Flow Accidents
- Locked Rotor Accident
- Control Rod Misalignment Accidents
- Turbine Trip Accident
- Steam Generator Tube Rupture Accident
- Rod Ejection Accident
- Steam Line Break Accidents
- Small Steam Line Break Accidents

The results of the reanalysis of these events are provided in Table 7-1 and are compared to the reference analysis Oconee 2 Cycle 18.

The radiological consequences (dose analysis) for the Chapter 15 accident analysis have been evaluated for the change in feed batch design. This evaluation concluded that the change in fuel design does not significantly impact the results of the analyses. Therefore, the conclusions from the reference analysis (Oconee 2 Cycle 18) are not impacted, and the resultant doses will remain within the post-accident acceptance criteria.

7.2 ECCS Analysis

LOCA analyses, applicable to the B&W designed Oconee Units 1, 2 and 3 operated by Duke Power Company, have been performed by Framatome Technologies Incorporated (FTI). The LOCA evaluation model, which has been approved by the NRC, is described in topical report BAW-10192P-A (Reference 18). The LOCA analyses comply with the criteria outlined in 10 CFR 50.46:

1. Peak cladding temperature (PCT) shall not exceed 2200 °F.
2. The percentage of local cladding oxidation shall not exceed 17%.
3. The maximum amount of hydrogen generated during the transient shall not exceed that which would be generated by the oxidation of 1% of the fuel cladding.
4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5. The mode of long term cooling shall be established.

In 1999, FTI has identified significant PCT increases associated with both reactor coolant pump (RCP) type and two-phase degradation models used in current Oconee RELAP5-based LOCA linear heat rate (LHR) licensing analyses (PSC 1-99). Duke notified the NRC of this LOCA error via Reference 19. FTI reanalyzed the LHR limits expected to have the most significant PCT increase (BOL Mark-B10 fuel) and found that the PCT could increase by 186 °F, with a maximum PCT of 2150 °F. For Oconee 1 Cycle 20, the LHR limits for the Mark-B10 fuel were reduced to restore PCT margin, and as a result, the limiting PCT for the Mark-B10F fuel is 2050 °F with reductions in LHR limits of between 0.3 kw/ft and 0.6 kw/ft. The final Mark-B10 LHR limits are provided in Table 7-2.

For the Mark-B11 fuel, separate analyses were performed to set the Mark-B11 LHR limits as shown in Table 7-3. These analyses included a mixed core evaluation to determine if a mixed core penalty was required during the transition to Mark-B11 fuel. The evaluation concluded that the small hydraulic differences between the Mark-B11 and Mark-B10 fuel assemblies did not create significantly different cross flow diversion. As a result, no PCT or LHR penalty is needed for the mixed core. The maximum PCT for the analysis of theMark-B11 fuel is 2037 °F.

In Reference 20, LHR adjustments were identified to show compliance with the large break LOCA Evaluation Model (EM). The LHR adjustments are used in the maneuvering analysis to ensue the limiting axial and radial peaking factors are bounded by the analyzed cases. The analyzed cases assume an axial peaking factor of 1.7 and the LHR adjustment provides a means to account for axial peaking factors that are significantly different than 1.7. For Oconee 1, Cycle 20 this approach is used in the maneuvering analysis to ensure that the SER requirements are met.

To address M5 cladding, changes to the FTI LOCA evaluation model were needed. Reference 12 provides these revised models which are used to analyze the Mark B-11 fuel (M5 cladding).

All LOCA results were calculated to be in conformance with the five criteria of 10 CFR 50.46, thus demonstrating acceptable results for the operation of Oconee 1 Cycle 20.

Table 7-1
Transient Analysis Results

Transient		Design Limit	O2C18 Mk-B10	Design Limit	O1C20 Mk-B11
Startup Accident	Peak Thermal Power (% FP)	75 (1)	73	75 (1)	77.6 (2)
	Peak RCS Pressure (psig)	2750	2747	2750	2746
	DNBR	1.5	N/A (3)	1.4	N/A (3)
Rod Withdrawl at Power	DNBR	1.5	1.719	1.4	1.870
	Peak RCS Pressure (psig)	2750	2611.5	2750	2608
Cold Water Accident	DNBR	1.5	N/A (3)	1.4	N/A (3)
	Peak Thermal Power (% FP)	100 (1)	96.7	100 (1)	97.8
	Peak RCS Pressure (psig)	2750	2165	2750	2170
Loss of Flow					
4/4 RCP Coastdown	DNBR	1.69	1.93	1.40	2.06
4/2 RCP Coastdown	DNBR	1.69	1.69 (4)	1.49	1.83
3/1 RCP Coastdown	DNBR	1.77	2.02	1.49	2.17
Locked Rotor					
4 RCP Initial Condition	Peak RCS Pressure (psig)	2750	2451.7	2750	2455.0
	DNBR	1.61	1.50 (4)	1.4	1.635
3 RCP Initial Condition	DNBR	1.62	1.33 (4)	1.4	1.579
Dropped Rod	Peak RCS Pressure (psig)	2750	N/A (7)	2750	N/A (7)
	DNBR	1.5	1.672	1.4	1.843
Turbine Trip	Peak RCS Pressure (psig)	2750	2614.1	2750	2611.8
	DNBR	1.5	N/A (3)	1.4	N/A (3)
Rod Ejection	Peak Enthalpy (cal/gm)	280	132.8	280	131.7
	Peak RCS Pressure (psig)	3000	2885	3000	2929
	Failed Fuel Fraction (%)	50 (5)	40.6	50 (5)	40.95
Large Steam Line Break					
w/ offsite power	DNBR	1.5	3.28	1.4	(6)
w/o offsite power	DNBR	1.5	1.45 (4)	1.4	1.805
Small Steam Line Break					
4 RCP Initial Condition	DNBR	1.5	1.443 (4)	1.4	1.605
3 RCP Initial Condition	DNBR	1.53	1.301 (4)	1.4	1.499

(1) permissible power
(2) Since the peak power exceeded the permissible power, DNB was evaluated and determined to be acceptable.
(3) The DNB acceptance criterion is not challenged during this event.
(4) The results of pin census analysis show that DNBR margin exists for all of the fuel rods.
(5) The actual acceptance criterion is the offsite dose needs to be within 10 CFR 100 criteria. The current dose analysis assumes 50% failed fuel.
(6) This case was not analyzed since the system response was similar to the reference analysis which demonstrated large margins.
(7) The accident response indicates that peak pressure will not be a concern for this event.

Table 7-2
LOCA LHR Limits Mark-B10

ELEVATION, ft	LOCA LHR LIMIT, kW/ft vs. BURNUP		
	0 MWd/mtU	30,000 MWd/mtU	62,000 MWd/mtU
0.000	15.6	15.6	11.6
2.506	16.5	16.5	11.6
4.264	16.8	16.8	11.6
6.021	17.0	17.0	11.6
7.779	17.0	17.0	11.6
9.536	16.7	16.7	11.6
12.00	15.8	15.8	11.6

Notes for Table 7-2:

1) The LHR limits presented above represent the power generated by the pin (i.e. all sources of useable energy caused by the fission process).
2) Linear interpolation for LHR limits is allowed between 30,000 MWd/mtU and 62,000 MWd/mtU.
3) The core endpoint (0 or 12 ft) LHR limits are 95% of the adjacent elevation between BOL and 30,000 MWd/mtU. At 62,000 MWd/mtU the endpoints are equivalent to the constant LHR limit values.
4) LHRs are valid for fuel enrichments between 3.0 and 5.0 $^{w}/_{o}$.
5) The BOL and MOL LOCA limits were calculated using a steady-state EDF of 0.973 for initial core energy deposition and a transient EDF of 1.0. The EOL LOCA limits were calculated using a steady-state EDF of 1.0 and a transient EDF of 1.1.

Table 7-3
LOCA LHR Limits Mark-B11

ELEVATION, ft	LOCA LHR LIMIT, kW/ft vs. BURNUP		
	0 MWd/mtU	40,000 MWd/mtU	62,000 MWd/mtU
0.000	15.5	15.5	12.6
2.506	16.3	16.3	12.6
4.264	16.5	16.5	12.6
6.021	16.8	16.8	12.6
7.779	16.5	16.5	12.6
9.536	16.2	16.2	12.6
12.00	15.4	15.4	12.6

Notes for Table 7-3:

1) The LHR limits presented above represent the power generated by the pin (i.e. all sources of useable energy caused by the fission process).
2) Linear interpolation for LHR limits is allowed between 40,000 MWd/mtU and 62,000 MWd/mtU.
3) The core endpoint (0 or 12 ft) LHR limits are 95% of the adjacent elevation between BOL and 40,000 MWd/mtU. At 62,000 MWd/mtU the endpoints are equivalent to the constant LHR limit values.
4) LHRs are valid for fuel enrichments 3.0 and 5.1 $^{w}/_{o}$.
5) The BOL and MOL LOCA limits were calculated using a steady-state EDF of 0.973 for initial core energy deposition and a transient EDF of 1.0. The EOL LOCA limits were calculated using a steady-state EDF of 1.0 and a transient EDF of 1.1.

8 PROPOSED CHANGES TO LICENSING BASIS DOCUMENTS

Revisions to the Technical Specifications, Core Operating Limits Report (COLR) and the Updated Final Safety Analysis Report (UFSAR) have been proposed for Oconee 1 Cycle 20 operation. These revisions reflect changes to accommodate the change in the feed batch fuel design. Table 8-1 lists the Technical Specification and Bases changes.

To implement Mark-B11 fuel, no COLR changes are required other than typical cycle-specific changes. For completeness, the Oconee Unit 1 Cycle 20 COLR is included in this submittal (Attachment VI).

The UFSAR changes resulting from the changes proposed for Oconee 1 Cycle 20 are not included in this submittal since the NRC has indicated that these changes are not reviewed. The update of the Oconee UFSAR will be implemented under 10 CFR 50.59 and submitted per the requirements of 10 CFR 50.71(e).

Table 8-1

Technical Specification Changes

Specification	Description of Change
TS 2.1.1.2	Add BWU correlation with its DNBR limit of 1.19.
TS 4.2.1	Add M5 as one of the cladding materials in the description of the fuel assemblies.
TS 5.6.5b	Add the M5 topical report (BAW-10227P) to the list of references. Revise the thermal hydraulic statistical core design topical (DPC-NE-2005-PA) to be Rev. 2.
B 2.1.1	Add the BWU correlation to analysis Mark-B11 fuel and its DNBR limit of 1.19. Add BAW-10199 PA to the list of references.
B 3.1.4 (A.2.4)	Revise the limits on ejected rod worth to be consistent with the updated analysis which includes Mark-B11 fuel.
B 3.4.1	Add the DNBR acceptance criterion of $\geq$ 1.19 for the BWU correlation.

9 REFERENCES

1. DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology," Duke Power Company, Oconee Nuclear Station, Rev. 1 (SER dated May 25, 1999).

2. Updated Final Safety Analysis Report, Oconee Nuclear Station, Duke Power Company.

3. Letter from J. W. Hampton (DEC) to Document Control Desk (NRC), Subject: Oconee Nuclear Station, Docket Nos. 50-269, -270, -287, Unit 3 Cycle 16 Reload Technical Specifications, Supplement 4, dated May 2, 1995.

4. Letter from H. N. Berkow (NRC) to M. S. Tuckman (DEC), Subject: Duke Power use of CROV Computer Code, dated June 19, 1995

5. BAW-10084P-A, Rev. 3, Program to determine In-reactor Performance of B&W Fuels – Cladding Creep Collapse, July 1991.

6. Letter from D. LaBarge (NRC) to W. R. McCollum, Jr. (DEC), Subject: Use of Framatome Cogema Fuels Topical Report on High Burnup – Oconee Nuclear Station, Units 1, 2, and 3, (TAC Nos. MA0405, MA0406, and MA0407), dated March 1, 1999.

7. Letter from M. S. Tuckman (DEC) to Document Control Desk (NRC), Subject: Duke Energy Corporation's use of FCF's Extended Burnup Evaluation Topical Report BAW-10186P-A, dated August 25, 1999.

8. BAW-10186P-A, Extended Burnup Evaluation, June 1997.

9. BAW-10179-A, Safety Criteria and Methodology for Acceptable Cycle Reload Analyses, August 1993.

10. DPC-NE-2008P-A, Duke Power Company Fuel Mechanical Reload Analysis Methodology using TACO3, April 1995.

11. Letter from J. H. Taylor (FCF) to Document Control Desk (NRC), Subject: Application of BAW-10186P-A, Extended Burnup Evaluation, dated October 28, 1997.

12. BAW-10227P, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, September 1997.

13. DPC-NE-1004A, Nuclear Design Methodology Using CASMO-3 / SIMULATE-3P, Duke Power Company, November 1992.

14. DPC-NE-2003P-A, Oconee Nuclear Stations Core Thermal- Hydraulic Methodology Using VIPRE-01, Duke Power Company, October 1989.

15. BAW-10143P-A, BWC Correlation of Critical Heat Flux, Babcock & Wilcox, April 1985.

16. DPC-NE-2005P-A, Rev. 2, Duke Power Company, Thermal-Hydraulic Statistical Core Design Methodology, June 1999.

17. BAW-10199P, The BWU Critical Heat Flux Correlations, Babcock and Wilcox, Addendum 1, SER dated April 6, 2000.

18. BAW-10192-PA, "BWNT LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," Rev. 0, (SER Dated February 18, 1997).

19. Letter from M. S. Tuckman (Duke) to USNRC, "Duke Energy Corporation, Oconee Nuclear Station, Units 1, 2 and 3, Docket Numbers 50-269, 50-270, and 50-287, Report Pursuant to 10CFR50.46, Error in LOCA Analysis," February 4, 1999.

20. Letter from M. S. Tuckman (Duke) to USNRC, "Duke Energy Corporation, Oconee Nuclear Station, Units 1, 2 and 3, Docket Numbers 50-269, 50-270, and 50-287, Report Pursuant to 10CFR50.46, Error Related to Application of the LBLOCA Evaluation Mocel," July 8, 1999.