

RELOAD ANALYSIS REPORT FOR WATERFORD 3 CYCLE 6

APRIL 1992

9204300366 920430
PDR ADDCK 05000382
P PDR

RELOAD ANALYSIS REPORT FOR WATERFORD 3 CYCLE 6

MAY 1992

TABLE OF CONTENTS

1. INTRODUCTION AND SUMMARY
 2. OPERATING HISTORY OF THE CURRENT CYCLE
 3. GENERAL DESCRIPTION
 4. FUEL SYSTEM DESIGN
 5. NUCLEAR DESIGN
 6. THERMAL-HYDRAULIC DESIGN
 7. NON-LOCA SAFETY ANALYSIS
 8. ECCS ANALYSIS
 9. REACTOR PROTECTION AND MONITORING SYSTEM
 10. TECHNICAL SPECIFICATIONS
 11. STARTUP TESTING
 12. REFERENCES
- APPENDIX A - EVALUATION OF CHANGE TO NUCLEAR DESIGN METHODS
- APPENDIX B - DEBPIS RESISTANT FUEL DESIGN

LIST OF TABLES

3-1	Cycle 6 Core Loading
5-1	Cycle 6 Nominal Physics Parameters
5-2	Cycle 6 Limiting Values of Reactivity Worths and Allowances for Hot Full Power Steam Line Break
5-3	Cycle 6 Reactivity Worth of CEA Regulating Groups at Hot Full Power
6-1	Cycle 6 Thermal Hydraulic Parameters at Full Power
7.0-1	Design Basis Events Considered in the Cycle 6 Safety Analysis
7.0-2	DBE's Evaluated with Respect to Offsite Dose Criterion
7.0-3	DBE's Evaluated with Respect to RCS Pressure Criterion
7.0-4	DBE's Evaluated with Respect to Fuel Performance
7.0-5	DBE's Evaluated with Respect to Shutdown Margin Criterion
7.0-6	Core Parameters Input to Safety Analysis
7.1.3-1	Sequence of Events for the Increased Main Steam Flow in Combination with a Loss of AC Power
8-1	Cycle 6 ECCS Analysis Significant System Parameters
8-2	Cycle 6 ECCS Analysis Significant Fuel Pin Parameters

LIST OF FIGURES

3-1	Cycle 6 Fresh Fuel
3-2	Cycle 6 Quarter Core Loading Pattern
3-3	Cycle 6 Assembly Averaged Burnup
3-4	CEA Bank Identification
3-5	In-Core Instrument Assemblies - Core Locations
5-1	PDIL for Regulating Groups
5-2	Part Length CEA Insertion Limit vs Thermal Power
5-3	Assembly Relative Power Density, HFP at BOC, Unrodded
5-4	Assembly Relative Power Density, HFP at MOC, Unrodded
5-5	Assembly Relative Power Density, HFP at EOC, Unrodded
5-6	Assembly Relative Power Density, HFP at BOC with PLCEA's
5-7	Assembly Relative Power Density, HFP at BOC with Bank 6
5-8	Assembly Relative Power Density, HFP at BOC with Bank 6 and PLCEA's
5-9	Assembly Relative Power Density, HFP at EOC with PLCEA's
5-10	Assembly Relative Power Density, HFP at EOC with Bank 6
5-11	Assembly Relative Power Density, HFP at EOC with Bank 6 and PLCEA's
8-1	ECCS Analysis, PCT versus Time for the Hot Spot Location

1.0 INTRODUCTION AND SUMMARY

This report provides an evaluation of the design and performance of Waterford Steam Electric Station Unit 3 during its sixth cycle of operation at 100% rated core power of 3390 MWt and NSSS power of 3410 MWt. Operating conditions for Cycle 6 are assumed to be consistent with those of previous Cycles and are summarized as full power operation under base load conditions. The core will consist of irradiated Batch C, F, and G assemblies, along with fresh Batch H assemblies. The Cycle 5 termination burnup has been assumed to be between 427 and 457 Effective Full Power Days (EFPD).

The safety criteria (margins of safety, dose limits, etc...) applicable to Waterford 3 were established in the FSAR (Reference 1-1) for Cycle 1. A review of all postulated accidents and anticipated operational occurrences (AOO's) was performed for Cycle 2, resulting in revisions to several safety analysis events, and for Cycles 3, 4, and 5, resulting in a negative 10CFR50.59 licensing finding. The FSAR as amended with information from the Cycle 2 Reload Analysis Report (References 1-2 and 1-3) constitutes the analyses of record for Waterford 3.

The Cycle 6 reload core characteristics have been evaluated with respect to the Reference Cycle. Specific differences in core fuel loadings have been accounted for in the present analysis. The Cycle 6 analysis results for postulated accidents and AOO's are summarized in Sections 7 and 8.

Two transients reanalyzed for Cycle 6 met NRC acceptance criteria, but exceeded the consequences determined in the Reference Cycle analyses documented in the Waterford 3 FSAR. These transients are (1) Large Break LOCA and (2) Excess Steam Demand with Loss of Offsite Power (also referred to as Excess Load).

Because the consequences of these two events exceeded that of the Reference Cycle, the Cycle 6 Reload Analysis Report is being submitted for NRC review. All other transients resulted in consequences which were bounded by the FSAR Reference Cycle analyses.

The Waterford 3 Cycle 6 reload required two methodology changes. The high burnup methodology of Reference 1-4 provides justification for peak rod burnups of up to 60,000 MWD/T; the peak rod burnup for Cycle 6 is 58,700 MWD/T. In addition, the fission gas release model used will be the NRC approved version of FATES3B (Reference 1-5).

The Cycle 6 reload required no Technical Specification changes.

Application of the Modified Statistical Combination of Uncertainties (MSCU) methodology (Reference 1-6) is anticipated for Cycle 6. Information presented in the following sections of this report will remain valid with the application of MSCU to Cycle 6, except that Reference 6-5 would be revised to the MSCU report, Reference 1-6. Application of the MSCU methodology would preserve the operating margins assumed in the Cycle 6 safety analyses.

2.0 OPERATING HISTORY OF THE CURRENT CYCLE

Waterford 3 is currently in its fifth fuel cycle which began with initial criticality on May 23, 1991. Full power operation was achieved on June 2, 1991.

It is presently estimated that Cycle 5 will terminate by September 18, 1992. The Cycle 5 termination point can vary between 427 EFPD and 457 EFPD to accommodate the plant schedule and still be within the burnup assumptions of the Cycle 6 analyses.

3.0 GENERAL DESCRIPTION

The Cycle 6 core will consist of those assembly types and numbers listed in Table 3-1. One Batch C1 assembly, forty-eight Batch E assemblies, and thirty-six Batch F assemblies will be removed from the Cycle 5 core to make way for 84 fresh Batch H assemblies plus one previously discharged C1 assembly. All eighty-four Batch G assemblies and forty-eight Batch F assemblies now in the core will be retained. One twice burned Batch C1 assembly discharged at the End of Cycle 2 will be reinserted.

The reload batch will consist of 8 type H0 assemblies, 20 type H1 assemblies with 8 burnable poison shims per assembly, and 56 type H2 assemblies with 16 burnable poison shims per assembly. These sub-batch types are zone-enriched and their configurations are shown in Figure 3-1.

The loading pattern for Cycle 6, showing fuel type and location in the quarter core, is displayed in Figure 3-2. The full core will be loaded with quarter core rotational symmetry.

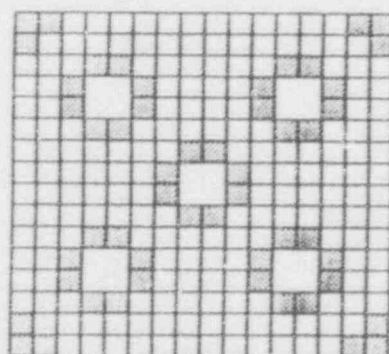
Figure 3-3 displays the beginning of Cycle 6 and end of Cycle 6 (490 EFPD) assembly average burnup distributions. These burnup distributions are based on a Cycle 5 length of 457 EFPD.

Control element assembly patterns and in-core instrument locations will remain unchanged from Cycle 5 and are shown in Figure 3-4 and Figure 3-5 respectively.

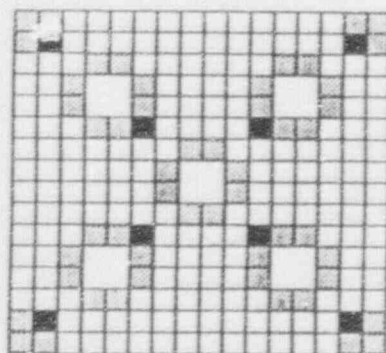
TABLE 3-1
WATERFORD 3
CYCLE 6 CORE LOADING

Assembly Design- ation	Number of Assemblies	Fuel Rods per Assembly	Initial Enrichment (w/o U-235)	Number Poison Rod Assembly	Initial Poison Loading (gm B ₁₀ /in)	Total Number of Fuel Poison Rods Rods	
D1	1	212 12	2.81 2.41	12	0.010	212 12	12
F0	16	184 52	4.05 3.65	0	0	2944 832	0
F1	20	176 52	4.05 3.65	8	0.016	3520 1040	160
F2	12	168 52	4.05 3.65	16	0.024	2016 624	192
G0	16	184 52	4.05 3.65	0	0	2944 832	0
G1	20	176 52	4.05 3.65	8	0.016	3520 1040	160
G2	48	168 52	4.05 3.65	16	0.024	8064 2496	768
H0	8	184 52	4.05 3.65	0	0	1472 416	0
H1	20	176 52	4.05 3.65	8	0.016	3520 1040	160
H2	56	168 52	4.05 3.65	16	0.024	9408 2912	896
Total	217					48864	2348

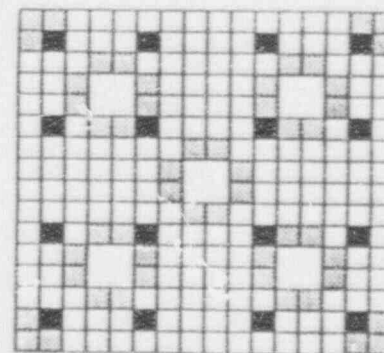
FIGURE 3-1
WATERFORD 3 CYCLE 6 FRESH FUEL



H0 FUEL



H1 FUEL



H2 FUEL

FUEL TYPE	No. OF ASSEMBLIES	ENRICHMENT		No. OF SHIMS	GM B-10/IN
		W/O □	W/O □		
H0	8	4.05	3.65	0	0.000
H1	20	4.05	3.65	8	0.016
H2	56	4.05	3.65	16	0.024

Figure 3-2
WATERFORD 3 CYCLE 6 QUARTER CORE LOADING PATTERN

X
A

A = Fuel Batch
X = QC Location (Cycle 6)

						1 F2	2 H0	
			3 F1	4 F1	5 G1	6 H1	7 G2	
		8 F1	9 H1	10 G0	11 H2	12 G2	13 H2	
	14 F2	15 H1	16 G2	17 H2	18 G1	19 H2	20 G2	
21 F1	22 H1	23 G2	24 G0	25 G2	26 H2	27 F0	28 H2	
29 F1	30 G0	31 H2	32 G2	33 G0	34 F0	35 H2	36 G2	
37 G1	38 H2	39 G1	40 H2	41 F0	42 H2	43 G2	44 H2	
45 F2	46 H1	47 G2	48 H2	49 F0	50 H2	51 G2	52 G1	53 G2
54 H0	55 G2	56 H2	57 G2	58 H1	59 G2	60 H2	61 G2	62 * C1

* REINSERTED FROM CYCLE 2

Figure 3-3
WATERFORD 3 CYCLE 6
Assembly Averaged Burnups
(EOC5 = 457 EFPD, EOC6 = 490 EFPD)

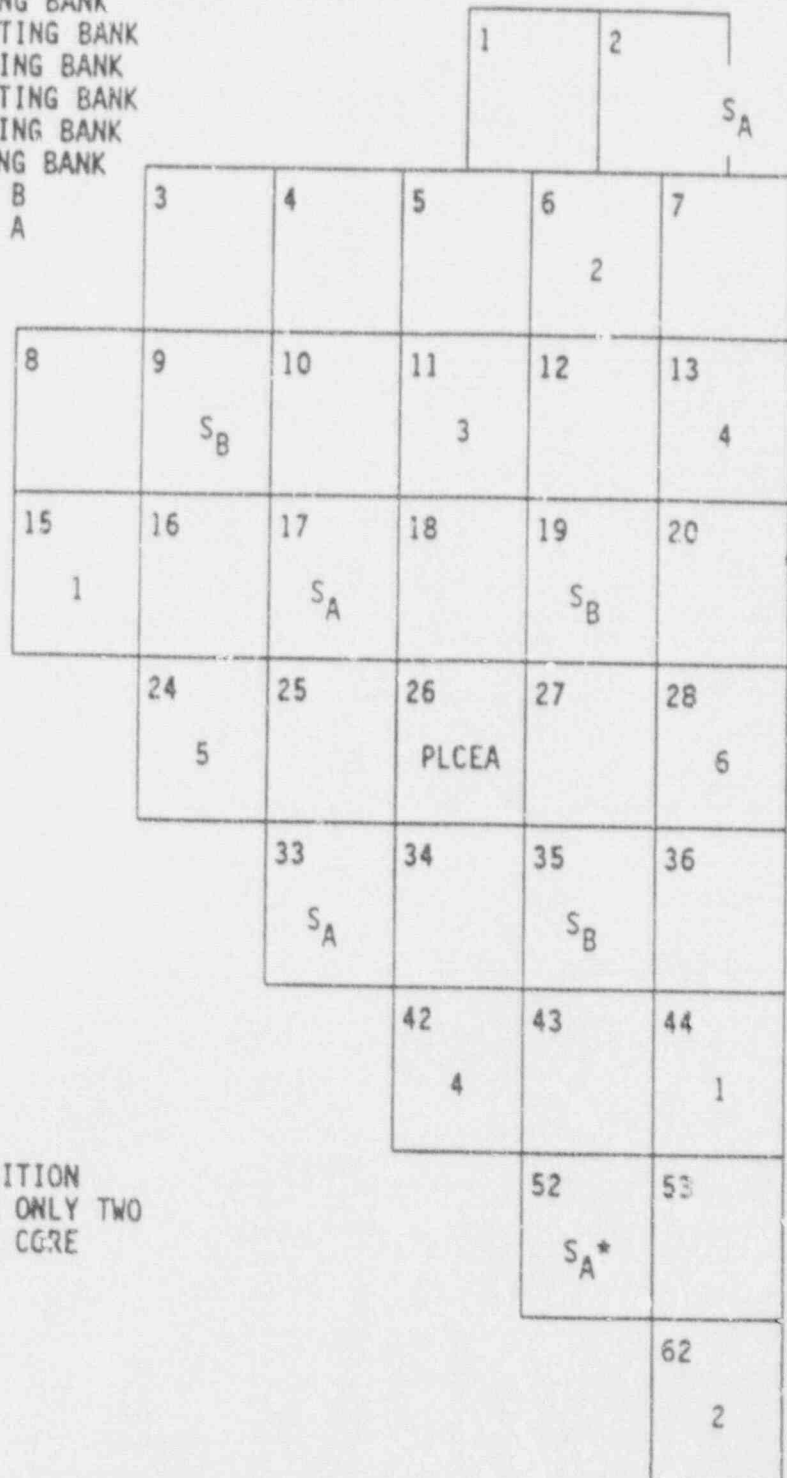
		1 F2		2 H0			
		37300		0			
		43900		14500			
		3 F1	4 F1	5 G1	6 H1	7 G2	
		35100	36900	18000	0	22300	
		40000	44300	31700	21400	41400	
		8 F1	9 H1	10 G0	11 H2	12 G2	13 H2
		36300	0	15100	0	22800	0
		42700	16300	32700	23100	43600	25300
		14 F2	15 H1	16 G2	17 H2	18 G1	19 H2
		39400	0	23200	0	17300	0
		45600	16900	41100	24400	39500	25700
		21 F1	22 H1	23 G2	24 G0	25 G2	26 H2
		35100	0	23200	12200	21300	0
		40000	16100	41100	33300	42500	25100
		29 F1	30 G0	31 H2	32 G2	33 G0	34 F0
		36900	15100	0	21300	12200	31900
		44300	32700	24400	42500	34000	50000
		37 G1	38 H2	39 G1	40 H2	41 F0	42 H2
		18000	0	17300	0	31900	0
		31700	23100	39500	25100	49900	25100
		45 F2	46 H1	47 G2	48 H2	49 F0	50 H2
		37300	0	22800	0	34400	0
		43900	21400	43600	25700	52200	25200
		54 H0	55 G2	56 H2	57 G2	58 H2	59 G2
		0	22200	0	21800	0	23100
		14500	41400	25300	43600	25200	44600
			56 H2	57 G2	58 H2	59 G2	60 H2
			0	21800	0	23100	0
			25300	43600	25200	44600	25500
							61 G2
							23100
							42200
							62 C1
							29000
							42800

X	Y
A	
B	

X = Quarter Core Assembly Number
Y = Fuel Batch
A = Assembly Average Burnup (MWD/T), BOC
B = Assembly Average Burnup (MWD/T), EOC

PLCEA - PART LENGTH CEA BANK

- 6 = LEAD REGULATING BANK
- 5 = SECOND REGULATING BANK
- 4 = THIRD REGULATING BANK
- 3 = FOURTH REGULATING BANK
- 2 = FIFTH REGULATING BANK
- 1 = LAST REGULATING BANK
- S_B = SHUTDOWN BANK B
- S_A = SHUTDOWN BANK A



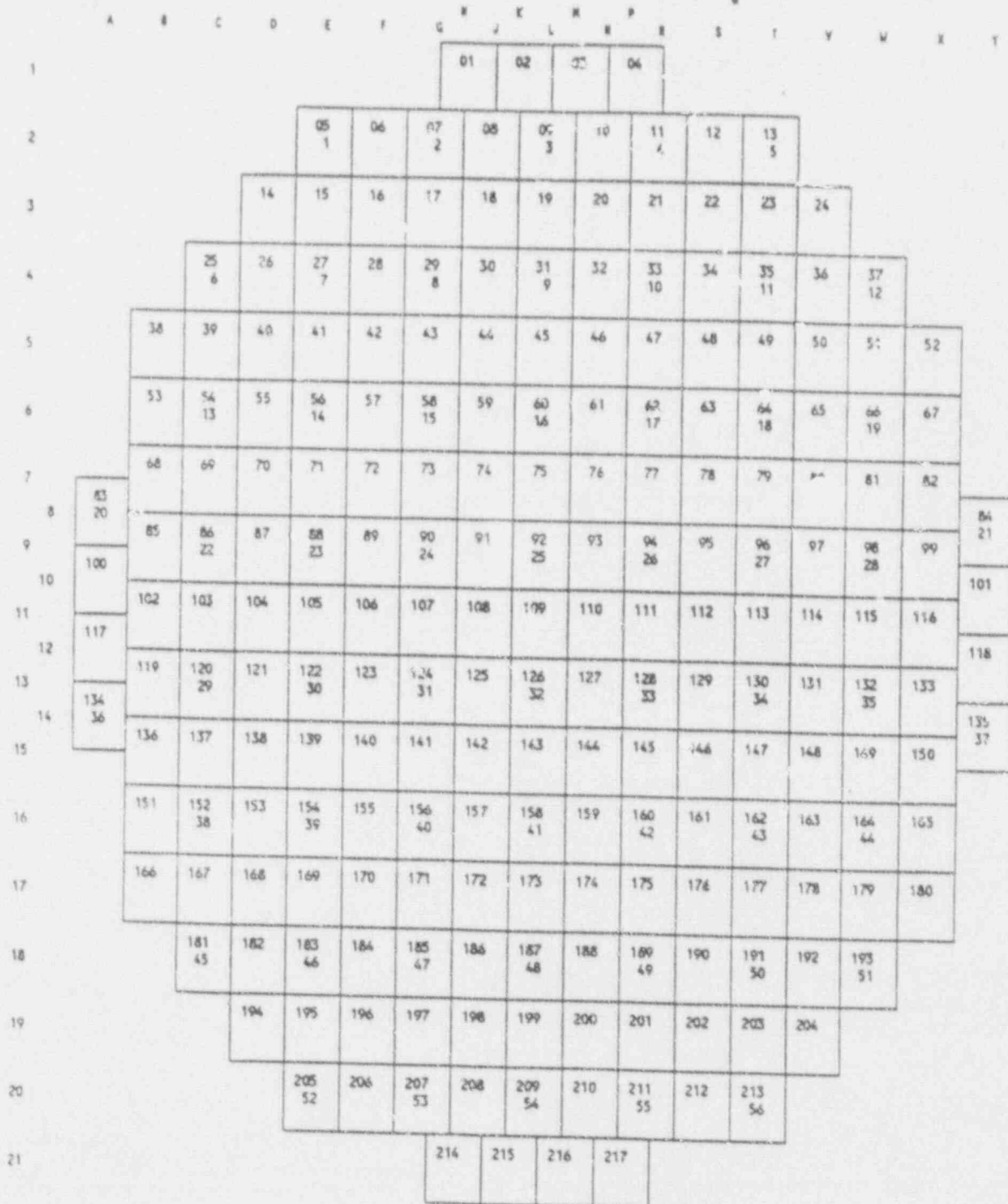
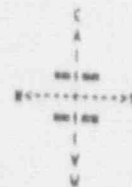
* SHUTDOWN ROD IN POSITION
53 IS AVAILABLE FOR ONLY TWO
DIAGONALLY OPPOSITE CORE
QUADRANTS.

LOUISIANA POWER & LIGHT CO. Waterford Steam Electric Station	WATERFORD 3 CEA BANK IDENTIFICATION	Figure 3-4
---	--	---------------

KEY TO MAP



XXX = Full Core Assembly Location
YYY = Instrument Assembly Number



LOUISIANA POWER & LIGHT CO. Waterford Steam Electric Station	WATERFORD 3 IN-CORE INSTRUMENT ASSEMBLIES CORE LOCATIONS	Figure 3-5
---	--	---------------

4.0 FUEL SYSTEM DESIGN

4.1 Mechanical Design

The mechanical design of the Waterford 3 Batch H reload fuel is essentially the same as the debris-resistant Batch G fuel that was introduced in Cycle 5. The changes associated with the debris-resistant design are described in Appendix B, which also describes the minor changes to the mechanical design since Cycle 2.

The primary differences between the Batch H and Batch G fuel designs are minor changes in the skirt that is located at the bottom of the upper end fitting's outer posts, and in the flange located at the upper end of the outer guide tube assembly. The changes improve the fabrication process. The outer post is screwed into the guide tube flange to secure the fuel assembly upper end fitting to the fuel assembly grid cage, and the post skirt is expanded into holes in the flange to prevent the post from unscrewing. These changes will minimize distortion of the guide posts during fabrication while still meeting the anti-rotation torque requirements. This design change is not expected to impact the in-reactor performance of the fuel.

4.2 Mitigation of Guide Tube Wear

All fuel assemblies in Cycle 6 will have stainless steel sleeves installed in the guide tubes to prevent guide tube wear.

4.3 Thermal Design

The thermal performance of composite fuel rods that envelope the fuel rods of the Batches present in Cycle 6 have been evaluated using the FATES3B version of the CE fuel evaluation model (References 4-3, 4-4, 4-5, and 4-6). The analysis was performed using a power history that enveloped the power and burnup levels representative of the peak fuel rod at each burnup interval, from Beginning of Cycle (BOC) to End of Cycle (EOC) burnups. The burnup range analyzed is in excess of that expected at the end of Cycle 6. Results of these burnup dependent fuel performance calculations were used in the Transient Analyses (Section 7) and in the Emergency Core Cooling System (ECCS) Analysis (Section 8) performed for Cycle 6.

4.4 Chemical Design

The material specifications that are used for the fuel cladding and the other fuel assembly components in Batch H fuel are the same as those that were used in Batch G fuel and essentially the same as those that were used for Batch D (Cycle 2) fuel.

4.5 Shoulder Gap Adequacy

All of the fuel assemblies that are to be used in Cycle 6 were evaluated to confirm that there is adequate shoulder gap clearance through the end of Cycle 6. The method used for Cycle 6 is that described in Reference 4-2. The initial shoulder gap in Batch F, Batch G, and Batch H fuel is 2.382 inches, the same as reported in the Cycle 2 Reload Analysis Report for Batch D. The

single Batch C assembly in the Cycle 6 core had an initial shoulder gap of 2.032 inches.

5.0 NUCLEAR DESIGN

5.1 Physics Characteristics

5.1.1 Fuel Management

The Cycle 6 core makes use of a very low-leakage fuel management scheme in which twice burned Batch F assemblies are placed on the core periphery. Most of the fresh Batch H assemblies are located throughout the interior of the core where they are mixed with other previously burned fuel in a pattern that minimizes power peaking. With this loading and a Cycle 5 endpoint at 441 EFPD, the Cycle 6 reactivity for full power operation is expected to be 462 EFPD (17,600 MWD/T). Explicit evaluations have been performed to assure applicability of all analyses to a Cycle 5 termination burnup of between 427 and 457 EFPD and for a Cycle 6 length up to 490 EFPD (18,600 MWD/T).

Characteristic physics parameters for cycle 6 are compared to those of the Reference Cycle in Table 5-1. The values in this table are intended to represent nominal core parameters. Those values used in the safety analyses contain appropriate uncertainties, or incorporate values to bound future operating cycles, and in all cases are conservative with respect to the values reported in Table 5-1.

Table 5-2 presents a summary of Control Element Assembly (CEA) reactivity worths and allowances for the end of Cycle 6 full power Main Steam Line Break (MSLB) transient with a comparison to the Reference Cycle data. The Cycle 6 values are explicitly calculated with 3D ROCS (Reference 5-2) while the Reference Cycle values were based on 2D POCS with adjustments for 3D effects. The full power steam line break was chosen to illustrate differences in CEA reactivity worths for the two cycles.

The CEA core locations and group identifications remain the same as in the Reference Cycle. The power dependent insertion limits (PDIL's) for regulating groups and part-length CEA groups are shown in Figures 5-1 and 5-2 respectively and remain the same as in the Reference Cycle. Table 5-3 shows the reactivity worths of various CEA groups calculated at power conditions for Cycle 6 and the Reference Cycle.

5.1.2 Power Distributions

Figures 5-3 through 5-5 illustrate the calculated All Rods Out (ARO) planar radial power distributions during Cycle 6. The one-pin planar radial power peaks presented in these figures are obtained from the middle region of the core. Time points at the beginning, middle, and end of cycle were chosen since the variation in maximum planar radial peak as a function of burnup is small.

Radial power distributions for selected rodged configurations are given for BOC and EOC in Figures 5-6 through 5-11. The rodged configurations shown are: part-length CEA's (PLCEA's); Bank 6; and Bank 6 plus the PLCEA's.

The radial power distributions described in this section are calculated data which do not include any uncertainties or allowances. The fine mesh calculations performed to determine these radial power peaks explicitly

account for augmented power peaking which is characteristics of fuel rods adjacent to the water holes.

Nominal axial peaking factors are expected to range from 1.16 at BOC to 1.07 at Middle of Cycle, to 1.10 at EOC.

5.1.3 Maximum Fuel Rod Burnup

The maximum fuel rod burnup of 58,700 MWD/T projected for the Waterford 3 Cycle 6 safety analysis is less than the 60,000 MWD/T limit presented in Reference 5-3. Reference 5-3 has been transmitted to the NRC by ABB/CE for generic approval. The physics data which are input to the Cycle 6 safety and fuel performance analyses are developed from explicit fine mesh calculations of fuel rod power and exposure.

Those physics data which are burnup dependent, for example, maximum fuel rod fluence and fuel rod power histories for FATES3B analyses, conservatively envelope core and fuel rod behavior at the maximum burnups as well as lower burnups. Also, the power levels of the high burnup rods are more than 30% below the EOC peak rod power levels.

5.2 Physics Analysis Methods

5.2.1 Analytical Input to In-Core Measurements

In-core detector measurement constants to be used in evaluating the reload cycle power distributions will be calculated in accordance with Reference 5-1.

5.2.2 Uncertainties in Measured Power Distributions

The planar radial power distribution measurement uncertainty of 6.92%, based upon Reference 5-1, will be applied to the Cycle 6 COLSS and CPC on-line calculations which use planar radial power peaks. The axial and three-dimensional power distribution measurement uncertainties are determined using the values in Reference 5-1 in conjunction with other monitoring and protection system measurement uncertainties.

5.2.3 Nuclear Design Methodology

As in the Reference Cycle, the Cycle 6 nuclear design was performed with two- and three-dimensional core models using the ROCS computer code and employing DIT calculated cross sections. The ROCS-DIT code and the MC module were described in Reference 5-2.

Recent developments in ABB/CE physics methodology (Reference 5-5) explicitly account for the following additional physical phenomena: (1) anisotropic scattering within the pin cells, (2) anisotropic neutron current at cell interfaces, (3) assembly discontinuity factors, and (4) the utilization of the Nodal Expansion Method (NEM) in ROCS instead of the previous Higher Order Difference (HOD) solution. The Cycle 6 design was performed using the Reference 5-5 methodology.

The Reference 5-5 methodology produces more accurate core power distribution predictions by improving: 1) the global radial power distribution, where power

sharing between neighboring assemblies is better modelled, and 2) the local fuel pin power distribution within an assembly, where the predicted distribution is flatter.

The ROCS-DIT methods and theories of Reference 5-2, as supplemented by Reference 5-5, have been reviewed by the NRC. In its acceptance of Reference 5-2, the NRC approved the general physics theories employed by ROCS-DIT and the analytical technique used to obtain the calculational uncertainties (95/95 limits) associated with the methodology, rather than the actual values of the uncertainties. ABB/CE has reviewed the available data base and has determined that the calculational uncertainties are not substantially different as a result of the new methods. The ROCS-DIT topical and its SER anticipated that improvements to the methodology would be made. The NRC, in its approval, only required CE to reevaluate the uncertainties associated with such analytical changes. Consequently, the ROCS-DIT Reference 5-5 methodology can be applied without additional NRC review. Additional details are provided in Appendix A to this report.

Negative reactivity insertion as a function of CEA scram bank position was calculated using the one dimensional space-time code FIESTA (Reference 5-4).

TABLE 5-1
WATERFORD 3 CYCLE 6
NOMINAL PHYSICS CHARACTERISTICS

<u>Dissolved Boron</u>	<u>Units</u>	<u>Reference Cycle</u>	<u>Cycle 6</u>
Dissolved Boron Concentration for Criticality, CEA's Withdrawn, Hot Full Power	PPM	1156	1110
<u>Boron Worth</u>			
Hot Full Power, BOC	PPM/% Δp	107	130
Hot Full Power, EOC	PPM/% Δp	85	96
<u>Moderator Temperature Coefficients</u>			
Hot Full Power, Equilibrium Xenon			
Beginning of Cycle	$10^{-4}\Delta p/^{\circ}F$	-0.1	-0.7
End of Cycle	$10^{-4}\Delta p/^{\circ}F$	-2.5	-3.1
<u>Doppler Coefficient</u>			
Hot Zero Power, EOC	$10^{-5}\Delta p/^{\circ}F$	-1.7	-1.7
Hot Full Power, BOC	$10^{-5}\Delta p/^{\circ}F$	-1.2	-1.2
Hot Full Power, EOC	$10^{-5}\Delta p/^{\circ}F$	-1.4	-1.4
<u>Total Delayed Neutron Fraction, β_{eff}</u>			
BOC	_____	0.0063	0.0062
EOC	_____	0.0051	0.0051
<u>Neutron Generation Time, l^*</u>			
BOC	10^{-6} sec	23.9	19.9
EOC	10^{-6} sec	30.0	26.6

TABLE 5-2

WATERFORD 3 CYCLE 6 LIMITING VALUES OF
 REACTIVITY WORTHS AND ALLOWANCES FOR HOT
 FULL POWER STEAM LINE BREAK, $\Delta\rho$, END-OF-CYCLE (EOC)

	Reference Cycle	Cycle 6
1. Worth of all CEAs Inserted	-12.1	-10.40
2. Stuck CEA Allowance	+ 2.3	+ 1.06
3. Worth of all CEAs less Highest Worth CEA Stuck Out	- 9.8	- 9.34
4. Full Power Dependent Insertion Limit CEA Bite	+ 0.3	+ 0.24
5. Calculated Scram Worth	- 9.5	- 9.1
6. Physics Uncertainty	+ 1.0	+ 0.57
7. Other Allowances	+ 0.2	+ 0.1
8. Net Available Scram Worth	- 8.3	- 8.43

TABLE S-3

WATERFORD 3 CYCLE 6
 REACTIVITY WORTH OF CEA REGULATING GROUPS
 AT HOT FULL POWER, $\Delta\rho$

Regulating CEA's	<u>Beginning of Cycle</u>		<u>End of Cycle</u>	
	<u>Reference Cycle</u>	<u>Cycle 6</u>	<u>Reference Cycle</u>	<u>Cycle 6</u>
Group 6	0.4	0.3	0.5	0.5
Group 5	0.5	0.5	0.6	0.4
Group 4	1.0	0.9	1.1	1.1

Note:

Values shown assume sequential group insertion.

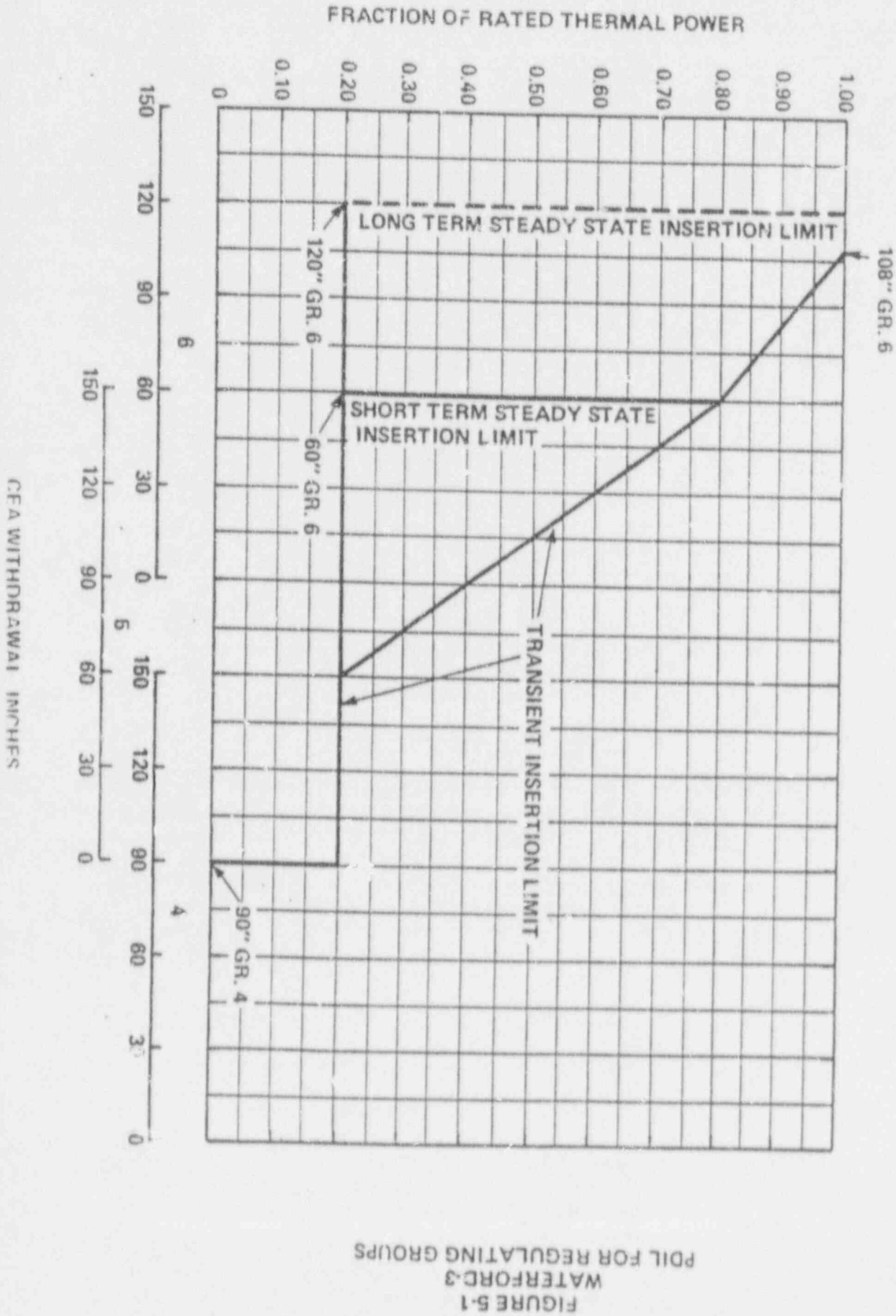
TABLE 5-3

WATERFORD 3 CYCLE 6
 REACTIVITY WORTH OF CEA REGULATING GROUPS
 AT HOT FULL POWER, $\Delta\rho$

Regulating CEAs	<u>Beginning of Cycle</u>		<u>End of Cycle</u>	
	Reference Cycle	Cycle 6	Reference Cycle	Cycle 6
Group 6	0.4	0.3	0.5	0.5
Group 5	0.5	0.5	0.6	0.4
Group 4	1.0	0.9	1.1	1.1

Note:

Values shown assume sequential group insertion.



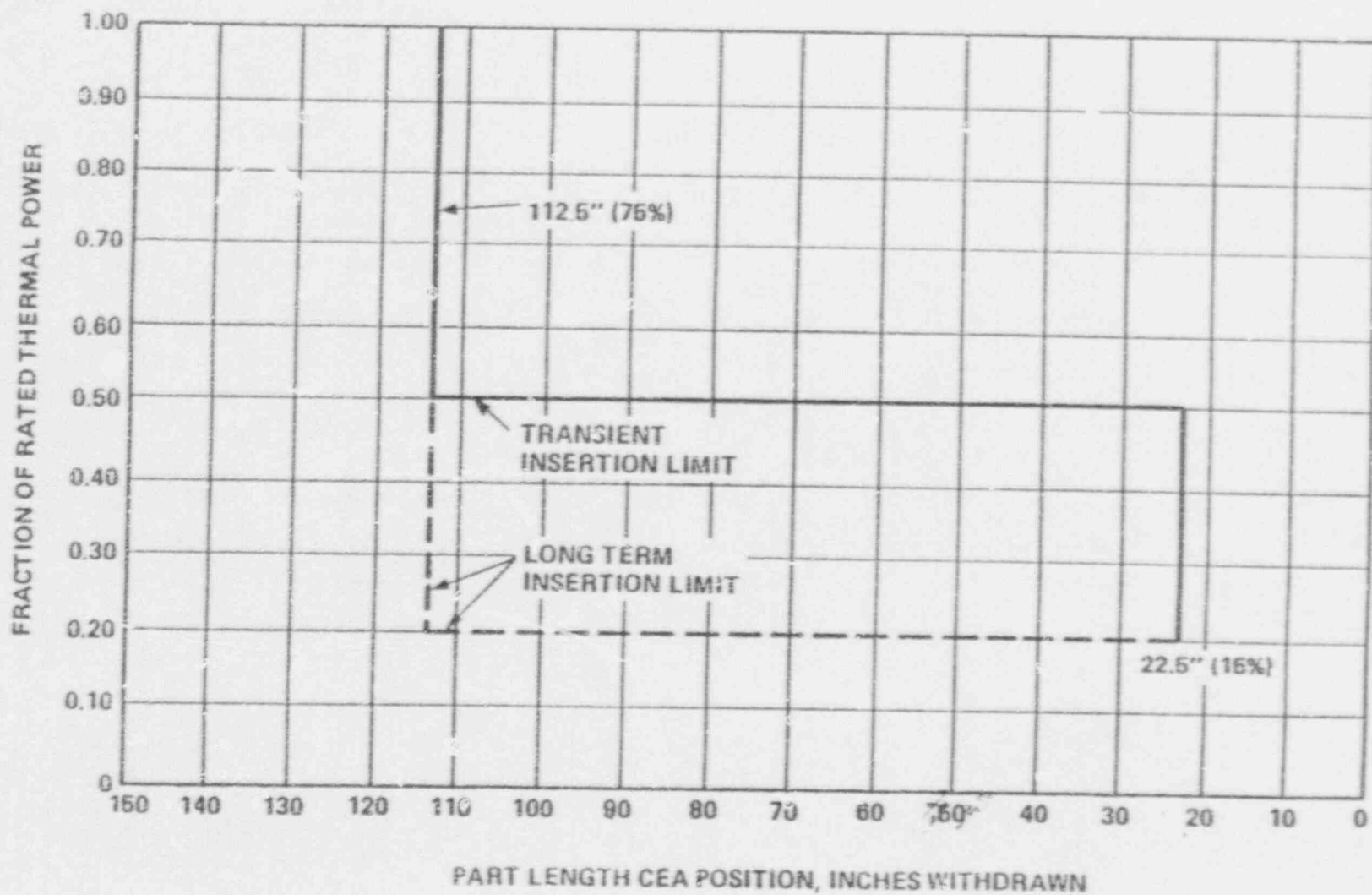


FIGURE 5.2
WATERFORD-3
PART LENGTH CEA INSERTION LIMIT vs THERMAL POWER

Figure 5-3

WATERFORD 3 CYCLE 6
 Assembly Relative Power Density
 HFP at BOC, Unrodded

		1 F2		2 H0				
		0.348		0.841				
		3 F1	4 F1	5 G1	6 H1	7 G2		
		0.237	0.371	0.720	1.175	1.095		
		8 F1	9 H1	10 G0	11 H2	12 G2	13 H2	
		0.324	0.852	0.953	1.180	1.127	1.303	
		14 F2	15 H1	16 G2	17 H2	18 G1	19 H2	20 G2
		0.311	0.913	1.025	1.301	1.211	1.290	1.139
21 F1	22 H1	23 G2	24 G0	25 G2	26 H2	27 F0	28 H2	
0.235	0.844	1.026	1.283	1.251	1.311	0.917	1.238	
29 F1	30 G0	31 H2	32 G2	33 G0	34 F0	35 H2	36 G2	
0.370	0.951	1.302	1.250	1.301	0.993	1.278	1.138	
37 G1	38 H2	39 G1	40 H2	41 F0	42 H2	43 G2	44 H2	
45 F2	0.719	1.180	1.210	1.307	0.989	1.321	1.224	1.340
0.348	46 H1	47 G2	48 H2	49 F0	50 H2	51 G2	52 G1	53 G2
54 H0	1.175	1.127	1.289	0.916	1.276	1.224	1.191	1.104
0.841	55 G2	56 H2	57 G2	58 H2	59 G2	60 H2	61 G2	62 C1
	1.095	1.303	1.139	1.238	1.138	1.340	1.104	0.769

Maximum 1-Pin Peak = 1.503 in Assembly 24

X	Y
Z	

X = Quarter Core Assembly Number
 Y = Fuel Batch
 Z = Integrated Power Density

Figure 5-4

WATERFORD 3 CYCLE 6
Assembly Relative Power Density
HFP at MOC, Unrodded

		1 F2 2 H0							
		0.351 0.771							
				3 F1	4 F1	5 G1	6 H1	7 G2	
				0.257	0.396	0.732	1.148	1.034	
		8 F1	9 H1	10 G0	11 H2	12 G2	13 H2		
		0.335	0.857	0.941	1.237	1.127	1.369		
		14 F2	15 H1	16 G2	17 H2	18 G1	19 H2	20 G2	
		0.323	0.890	0.957	1.306	1.197	1.388	1.184	
		21 F1	22 H1	23 G2	24 G0	25 G2	26 H2	27 F0	28 H2
		0.255	0.851	0.957	1.132	1.143	1.354	0.969	1.364
		29 F1	30 G0	31 H2	32 G2	33 G0	34 F0	35 H2	36 G2
		0.395	0.939	1.306	1.142	1.174	0.978	1.366	1.170
		37 G1	38 H2	39 G1	40 H2	41 F0	42 H2	43 G2	44 H2
45 F2	0.732	1.237	1.197	1.351	0.976	1.361	1.207	1.386	
0.351	46 H1	47 G2	48 H2	49 F0	50 H2	51 G2	52 G1	53 G2	
54 H0	1.148	1.126	1.388	0.968	1.365	1.207	1.119	1.035	
0.771	55 G2	56 H2	57 G2	58 H2	59 G2	60 H2	61 G2	62 C1	
	1.034	1.369	1.184	1.364	1.170	1.386	1.035	0.739	

Maximum 1-Pin Peak = 1.485 in Assembly 19

X	Y
Z	

X = Quarter Core Assembly Number
Y = Fuel Batch
Z = Integrated Power Density

WATERFORD 3 CYCLE 6
Assembly Relative Power Density
HFP at EOC, Unrodded

Maximum 1-Pin Peak = 1.488 in Assembly 48

X = Quarter Core Assembly Number
Y = Fuel Batch
Z = Integrated Power Density

Figure 5-6

WATERFORD 3 CYCLE 6
Assembly Relative Power Density
HFP at BOC with PLCEA's

		1 F2		2 H0				
		0.355		0.837				
		3 F1	4 F1	5 G1	6 H1	7 G2		
		0.244	0.382	0.733	1.176	1.112		
		8 F1	9 H1	10 G0	11 H2	12 G2	13 H2	
		0.333	0.858	0.966	1.177	1.142	1.302	
		14 F2	15 H1	16 G2	17 H2	18 G1	19 H2	20 G2
		0.319	0.921	1.048	1.290	1.198	1.272	1.148
21 F1	22 H1	23 G2	24 G0	25 G2	26 H2	27 F0	28 H2	
0.242	0.850	1.048	1.298	1.246	1.191	0.909	1.227	
29 F1	30 G0	31 H2	32 G2	33 G0	34 F0	35 H2	36 G2	
0.381	0.963	1.290	1.245	1.295	0.989	1.266	1.154	
37 G1	38 H2	39 G1	40 H2	41 F0	42 H2	43 G2	44 H2	
45 F2	0.733	1.176	1.197	1.187	0.985	1.318	1.246	1.352
0.355	46 H1	47 G2	48 H2	49 F0	50 H2	51 G2	52 G1	53 G2
54 H0	1.176	1.142	1.271	0.908	1.265	1.245	1.229	1.148
0.838	55 G2	56 H2	57 G2	58 H2	59 G2	60 H2	61 G2	62 C1
	1.112	1.302	1.148	1.227	1.154	1.352	1.148	0.789

Maximum 1-Pin Peak = 1.513 in Assembly 24

X	Y
	Z

X = Quarter Core Assembly Number
Y = Fuel Batch
Z = Integrated Power Density

Figure 5-7

WATERFORD 3 CYCLE 6
Assembly Relative Power Density
HFP at BOC with Bank 6

		1 F2		2 H0				
		0.364		0.856				
		3 F1	4 F1	5 G1	6 H1	7 G2		
		0.260	0.401	0.754	1.194	1.124		
		8 F1	9 H1	10 G0	11 H2	12 G2	13 H2	
		0.357	0.914	1.015	1.208	1.136	1.274	
		14 F2	15 H1	16 G2	17 H2	18 G1	19 H2	20 G2
		0.342	0.987	1.362	1.233	1.209	1.014	
		21 F1	22 H1	23 G2	24 G0	25 G2	26 H2	27 F0
		0.257	0.905	1.117	1.381	1.323	1.304	0.816
		29 F1	30 G0	31 H2	32 G2	33 G0	34 F0	35 H2
		0.399	1.012	1.363	1.322	1.356	1.003	1.183
		37 G1	38 H2	39 G1	40 H2	41 F0	42 H2	43 G2
		0.753	1.206	1.232	1.300	0.999	1.306	1.205
45 F2	0.364	46 H1	47 G2	48 H2	49 F0	50 H2	51 G2	52 G1
		1.194	1.135	1.208	0.815	1.182	1.205	1.199
54 H0	0.856	55 G2	56 H2	57 G2	58 H2	59 G2	60 H2	61 G2
		1.124	1.274	1.014	0.724	1.005	1.291	1.119
								62 C1
								0.772

Maximum 1-Pin Peak = 1.623 in Assembly 24

X	Y
Z	

X = Quarter Core Assembly Number
Y = Fuel Batch
Z = Integrated Power Density

Figure 5-8

WATERFORD 3 CYCLE 5
 Assembly Relative Power Density
 HFP at BOC, Bank 6 and PLCEA's

		1 F2 2 H0						
		0.376 0.885						
		3 F1	4 F1	5 G1	6 H1	7 G2		
		0.264	0.408	0.770	1.226	1.156		
		8 F1	9 H1	10 G0	11 H2	12 G2	13 H2	
		0.362	0.924	1.023	1.220	1.155	1.299	
		14 F2	15 H1	16 G2	17 H2	18 G1	19 H2	20 G2
		0.346	0.996	1.122	1.355	1.216	1.208	1.022
		21 F1	22 H1	23 G2	24 G0	25 G2	26 H2	27 F0 28 H2
		0.262	0.915	1.121	1.374	1.296	1.187	0.800 0.722
		29 F1	30 G0	31 H2	32 G2	33 G0	34 F0	35 H2 36 G2
		0.406	1.021	1.356	1.295	1.326	0.981	1.175 1.007
		37 G1	38 H2	39 G1	40 H2	41 F0	42 H2	43 G2 44 H2
		0.770	1.219	1.215	1.183	0.977	1.303	1.214 1.305
45 F2	0.376	46 H1	47 G2	48 H2	49 F0	50 H2	51 G2	52 G1 53 G2
		1.226	1.154	1.207	0.799	1.174	1.214	1.217 1.139
54 H0	0.885	55 G2	56 H2	57 G2	58 H2	59 G2	60 H2	61 G2 62 C1
		1.156	1.299	1.022	0.722	1.007	1.305	1.139 0.738

Maximum 1-Pin Peak = 1.602 in Assembly 24

X	Y
Z	

X = Quarter Core Assembly Number
 Y = Fuel Batch
 Z = Integrated Power Density

WATERFORD 3 CYCLE 6
Assembly Relative Power Density
HFP at EOC with PLCEA's

Maximum 1-Pin Peak = 1.506 in Assembly 56

X = Quarter Core Assembly Number
Y = Fuel Batch
Z = Integrated Power Density

Figure 5-10

WATERFORD 3 CYCLE 6
Assembly Relative Power Density
HFP at EOC with Bank 6

		1 F2		2 H0													
		0.417		0.825													
		3 F1		4 F1		5 G1		6 H1		7 G2							
		0.353		0.503		0.831		1.197		1.049							
		8 F1		9 H1		10 G0		11 H2		12 G2		13 H2					
		0.448		1.042		1.057		1.358		1.120		1.363					
		14 F2		15 H1		16 G2		17 H2		18 G1		19 H2		20 G2			
		0.435		1.068		1.057		1.412		1.188		1.312		1.009			
		21 F1		22 H1		23 G2		24 G0		25 G2		26 H2		27 F0		28 H2	
		0.351		1.037		1.056		1.144		1.122		1.334		0.844		0.762	
		29 F1		30 G0		31 H2		32 G2		33 G0		34 F0		35 H2		36 G2	
		0.502		1.057		1.412		1.122		1.103		0.935		1.230		0.947	
		37 G1		38 H2		39 G1		40 H2		41 F0		42 H2		43 G2		44 H2	
45 F2		0.831		1.358		1.188		1.333		0.934		1.288		1.071		1.252	
0.417		46 H1		47 G2		48 H2		49 F0		50 H2		51 G2		52 G1		53 G2	
54 H0		1.198		1.120		1.312		0.843		1.229		1.071		0.990		0.928	
0.825		55 G2		56 H2		57 G2		58 H2		59 G2		60 H2		61 G2		62 C1	
		1.049		1.363		1.009		0.762		0.947		1.252		0.928		0.703	

Maximum 1-Pin Peak = 1.561 in Assembly 31

X	Y
Z	

X = Quarter Core Assembly Number
Y = Fuel Batch
Z = Integrated Power Density

Figure 5-11

WATERFORD 3 CYCLE 6
Assembly Relative Power Density
HFP at EOC, Bank 6 and PLCEA's

		1 F2		2 H0				
		0.431		0.853				
		3 F1	4 F1	5 G1	6 H1	7 G2		
		0.361	0.514	0.851	1.229	1.079		
		8 F1	9 H1	10 G0	11 H2	12 G2	13 H2	
		0.457	1.058	1.070	1.374	1.139	1.389	
		14 F2	15 H1	16 G2	17 H2	18 G1	19 H2	20 G2
		0.444	1.083	1.065	1.406	1.172	1.309	1.016
21 F1	22 H1	23 G2	24 G0	25 G2	26 H2	27 F0	28 H2	
0.359	1.052	1.064	1.141	1.095	1.209	0.824	0.757	
29 F1	30 G0	31 H2	32 G2	33 G0	34 F0	35 H2	36 G2	
0.513	1.069	1.403	1.095	1.073	0.910	1.216	0.947	
37 G1	38 H2	39 G1	40 H2	41 F0	42 H2	43 G2	44 H2	
45 F2	0.851	1.374	1.172	1.208	0.908	1.200	1.076	1.263
0.431	46 H1	47 G2	48 H2	49 F0	50 H2	51 G2	52 G1	53 G2
54 H0	1.230	1.139	1.309	0.824	1.216	1.076	1.003	0.943
0.853	55 G2	56 H2	57 G2	58 H2	59 G2	60 H2	61 G2	62 C1
	1.079	1.389	1.016	0.757	0.947	1.263	0.943	0.717

Maximum 1-Pin Peak = 1.564 in Assembly 31

X	Y
Z	

X = Quarter Core Assembly Number
Y = Fuel Batch
Z = Integrated Power Density

6.0 THERMAL-HYDRAULIC DESIGN

6.1 DNBR Analysis

Steady State Departure from Nucleate Boiling Ratio (DNBR) analyses of Cycle 6 at the core rated power level of 3390 MWt have been performed using the TORC computer code described in Reference 6-1, the CE-1 Critical Heat Flux (CHF) correlation described in Reference 6-2, the simplified TORC modeling methods described in Reference 6-3, and the CETOP code described in Reference 6-4.

Table 6-1 presents a comparison of pertinent thermal-hydraulic design parameters for Cycle 6 and the Reference Cycle. The Statistical Combination of Uncertainties (SCU) methodology presented in Reference 6-5 was applied with Waterford 3 specific data. This was done using the calculational factors listed in Table 6-1 and other uncertainty factors at the 95/95 confidence/probability level to define a design limit of 1.26 on CE-1 minimum DNBR.

The Cycle 6 DNBR limit includes the following allowances:

1. NRC specified allowances for TORC code uncertainty and CE-1 CHF correlation cross validation uncertainty, as discussed in Reference 6-10.
2. An NRC imposed 0.01 DNBR penalty for MID-1 grids as discussed in Reference 6-6, 6-7, and 6-8.
3. Rod bow penalty as discussed in Section 6.2 below.

6.2 Effects of Fuel Rod Bowing on DNBR Margin

Effects of fuel rod bowing on DNBR margin have been incorporated in the safety and setpoint analyses in the manner discussed in References 6-5 and 6-9. The penalty used for this analysis, 1.75% on minimum DNBR, is valid for bundle burnups up to 30,000 MWD/T. This penalty is included in the 1.26 DNBR limit. For assemblies with burnups greater than 30,000 MWD/T, sufficient margin exists to offset rod bow penalties due to the lower radial power peaks in these higher burnup batches. Hence, the rod bow penalty based upon Reference 6-9 for 30,000 MWD/T is applicable for all assembly burnups expected for Cycle 6.

TABLE 6-1

Waterford 3 Cycle 6
Thermal Hydraulic Parameters at Full Power

<u>General Characteristics</u>	<u>Units</u>	<u>Reference Cycle (2)</u>	<u>Cycle 6</u>
Total Heat Output (Core only)	MWt	3390	3390
	10^6 BTU/hr	11570	11570
Fraction of Heat Generated in Fuel Rod	--	0.975	0.975
Primary System Pressure (Nominal)	psia	2250	2250
Inlet Temperature (Nominal)	$^{\circ}$ F	553.0	553.0
Total Reactor Coolant Flow (Minimum Steady State)	gpm	396,000	396,000
	10^6 lb/hr	148.0	148.0
Coolant Flow Through Core (Minimum)	10^6 lb/hr	144.2	144.2
Hydraulic Diameter (Nominal Channel)	ft	0.039	0.039
Average Mass Velocity	10^6 lb/hr-ft ²	2.64	2.64
Pressure Drop Across Core (Minimum steady state flow irreversible ΔP over entire fuel assembly)	psi	15.4	15.4
Total Pressure Across Vessel (Based on nominal dimensions and minimum steady state flow)	psi	41.5	41.5
Core Average Heat Flux (Accounts for fraction of heat generated in fuel rod and axial densification factor)	BTU/hr-ft ²	182,700***	185,400*
Total Heat Transfer Area (Accounts for axial densification factor)	ft ²	61,700***	60,800*

Table 6-1 (continued)

<u>General Characteristics</u>	<u>Units</u>	<u>Reference Cycle (2)</u>	<u>Cycle 6</u>
Film Coefficient at Average Conditions	BTU/hr-ft ² -°F	6200	6200
Average Film Temperature Difference	°F	29.3	29.9
Average Linear Heat Rate of Undensified Fuel Rod (Accounts for fraction of heat generated in fuel rod)	kW/ft	5.34***	5.42*
Average Core Enthalpy Rise	BTU/lb	80.3	80.3
Maximum Clad Surface Temperature	°F	656.7	656.7
Engineering Heat Flux Factor	----	1.03**	1.03**
Engineering Factor on Hot Channel Heat Input	----	1.03**	1.03**
Rod Pitch, Bowing and Clad Diameter Factor	----	1.05**	1.05**
Fuel Densification Factor (Axial)	----	1.002	1.00

NOTES:

- * Based on 2348 poison rods.
- ** These factors have been combined statistically with other uncertainty factors at the 95/95 confidence/probability level to define a new design limit on CE-1 minimum DNBR when iterating on power as discussed in Reference 6-5.
- *** Based on 1728 poison rods.

7.0 NON-LOCA SAFETY ANALYSIS

7.0.1 Introduction

This section presents the results of the Waterford 3 Cycle 6 Non-LOCA safety analyses at a rated power of 3410 MWt.

The Design Basis Events (DBE's) considered in the safety analyses are listed in Table 7.0.1. These events are categorized into three groups: Moderate Frequency, Infrequent, and Limiting Fault Events. For the purpose of this report, the Moderate Frequency and Infrequent Events are referred to as Anticipated Operational Occurrences (AOO's). The DBE's were evaluated with respect to four criteria: Offsite Dose, Reactor Coolant System Pressure, Fuel Performance (DNBR and Fuel Centerline Melt Specified Acceptable Fuel Design Limits (SAFDL's)), and Loss of Shutdown Margin. Tables 7.0-2 and 7.0-5 present the list of events analyzed for each criterion. All events were reevaluated to assure that they meet their respective criteria for Cycle 6. The DBE's chosen for analysis for each criterion are the limiting events with respect to that criterion.

For the event analyses presented, a discussion of the reason(s) for the reanalysis, a discussion of the cause(s) of the event, a description of the analyses performed, results, and conclusions are included.

7.0.2 Methods of Analysis

The analytical methodology used for Waterford 3 Cycle 6 non-LOCA safety analyses is the same as previously presented unless otherwise stated in the event presentation. Only the methodology that has been previously reviewed and approved on the Waterford 3 docket or on other dockets is used. The Reference Cycle for the individual DBE's is taken to be the last Cycle that the results for the individual events were presented to the NRC (Cycle 1, Reference 7-1, or Cycle 2, Reference 7-2).

Changes in inputs to the non-LOCA safety analyses, whether from the Cycle 6 core loading or from other changes to the plant configuration are assessed to determine if a reanalysis of any DBE is required.

7.0.3 Mathematical Models

The following mathematical models and computer codes were used to analyze the DBE's for Waterford 3 Cycle 6.

Plant response for non-LOCA events was simulated using the CESEC III computer code (Reference 7-3). Simulation of the fluid conditions within the hot channel of the reactor core and calculation of DNBR was performed using the CETOP-D computer code described in Reference 7-6.

The HERMITE computer code (Reference 7-7) was used to simulate the reactor core for analyses which required more spatial detail than is provided by a point kinetics model.

The TORC computer code (References 7-8 and 7-9) is used to simulate the fluid conditions within the reactor core and to calculate fuel pin DNBR for the sheared shaft event.

Determination of DNBR for the post-trip return-to-power portion of the Main Steam Line event is based on the correlation developed by R. V. Macbeth (Reference 7-4) with corrections to account for non-uniform axial heat flux developed by Lee (Reference 7-5). This methodology is the same as that employed in the Reference Cycle analysis.

The number of fuel pins predicted to experience clad failure is taken as the number of pins which have a CE-1 DNBR value below 1.26, except for analyses in which the method of statistical convolution (Reference 7-10) has been presented to the NRC on the Waterford 3 docket.

7.0.4 Input Parameters and Analysis Assumptions

Table 7.0-6 summarizes the core parameters assumed in the Cycle 6 transient analysis and compares them to the values used in the previous Cycle (Reference 7-11). Specific initial conditions for each event are tabulated in the section of the report summarizing that event. For some of the DBE's, certain initial core parameters were assumed to be more limiting than the actual calculated Cycle 6 values (e.g., CEA worth at trip, moderator temperature coefficient).

7.0.5 Conclusion

All DBE's were evaluated for Waterford 3 Cycle 6 to determine whether their results are bounded by the Reference Cycle. It was determined that the consequences of all Cycle 6 non-LOCA transients are bounded by the results already on the Waterford 3 docket except for the Increased Main Steam Flow with Loss of Offsite Power event. A discussion of this event for Cycle 6 is included in Section 7.1.3.

Table 7.0-1

Waterford 3

Design Basis Events Considered in the Cycle 6 Safety Analysis

- 7.1 Increase in Heat Removal by the Secondary System
 - 7.1.1 Decrease in Feedwater Temperature
 - 7.1.2 Increase in Feedwater Flow
 - 7.1.3 Increased Main Steam Flow
 - 7.1.4 Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve
 - 7.1.5 * Steam System Piping Failures
- 7.2 Decrease in Heat Removal by the Secondary System
 - 7.2.1 Loss of External Load
 - 7.2.2 Turbine Trip
 - 7.2.3 Loss of Condenser Vacuum
 - 7.2.4 Loss of Normal AC Power
 - 7.2.5 Loss of Normal Feedwater
 - 7.2.6 * Feedwater System Pipe Leaks
- 7.3 Decrease in Reactor Coolant Flowrate
 - 7.3.1 Partial Loss of Forced Reactor Coolant Flow
 - 7.3.2 Total Loss of Forced Reactor Coolant Flow
 - 7.3.3 * Single Reactor Coolant Pump Shaft Seizure / Sheared Shaft
- 7.4 Reactivity and Power Distribution Anomalies
 - 7.4.1 Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition
 - 7.4.2 Uncontrolled CEA Withdrawal at Power
 - 7.4.3 CEA Misoperation Events
 - 7.4.4 Chemical and Volume Control System (CVCS) Malfunction (Inadvertent Boron Dilution)
 - 7.4.5 Startup of an Inactive Reactor Coolant System Pump
 - 7.4.6 * CEA Ejection
- 7.5 Increase in Reactor Coolant System Inventory
 - 7.5.1 CVCS Malfunction
 - 7.5.2 Inadvertent Operation of the ECCS During Power Operation
- 7.6 Decrease in Reactor Coolant System Inventory
 - 7.6.1 Pressurizer Pressure Decrease Events
 - 7.6.2 * Small Primary Line Break Outside Containment
 - 7.6.3 * Steam Generator Tube Rupture
- 7.7 Miscellaneous
 - 7.7.1 Asymmetric Steam Generator Events

* Categorized as Limiting Fault Events

Table 7.0-2

DBE's Evaluated with Respect to Offsite Dose Criterion

<u>Section</u>	<u>Event</u>	<u>Results</u>
<i>A) Anticipated Operational Occurrences</i>		
7.1.4	1) Unadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve	Bounded by Reference Cycle
7.2.4	2) Loss of Normal AC Power	Bounded by Reference Cycle
<i>B) Limiting Fault Events</i>		
7.1.5.a	1) Steam System Piping Failures: a) Pre-Trip Power Excursions	Bounded by Reference Cycle
7.1.5.b	b) Post Trip Analysis	Bounded by Reference Cycle
7.2.6	2) Feedwater System Pipe Breaks	Bounded by Reference Cycle
7.3.3	3) Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft	Bounded by Reference Cycle
7.6.2	4) Small Primary Line Break Outside Containment	Bounded by Reference Cycle
7.6.3	5) Steam Generator Tube Rupture	Bounded by Reference Cycle

Table 7.0-3

DBE's Evaluated with Respect to LCS Pressure Criterion

<u>Section</u>	<u>Event</u>	<u>Results</u>
<i>A) Anticipated Operational Occurrences</i>		
7.2.1	1) Loss of External Load	Bounded by Reference Cycle
7.2.2	2) Turbine Trip	Bounded by Reference Cycle
7.2.3	3) Loss of Condenser Vacuum	Bounded by Reference Cycle
7.2.4	4) Loss of Normal AC Power	Bounded by Reference Cycle
7.2.5	5) Loss of Normal Feedwater	Bounded by Reference Cycle
7.4.1	6) Uncontrolled CEA Withdrawal from Subcritical or Low Power Conditions	Bounded by Reference Cycle
7.4.2	7) Uncontrolled CEA Withdrawal at Power	Bounded by Reference Cycle
7.5.1	8) CVCS Malfunction	Bounded by Reference Cycle
7.5.2	9) Inadvertent Operation of the ECCS During Power Operation	Bounded by Reference Cycle
<i>B) Limiting Fault Events</i>		
7.2.6	1) Feedwater System Pipe Breaks	Bounded by Reference Cycle

Table 7.0-4

DBE's Evaluated with Respect to Fuel Performance

Section	Event	Results
A) Anticipated Operational Occurrences		
7.1.1	1) Decrease in Feedwater Temperature	Bounded by Reference Cycle
7.1.2	2) Increase in Feedwater Flow	Bounded by Reference Cycle
7.1.3	3) Increase in Main Steam Flow	Not Bounded by Reference Cycle (presented)
7.1.4	4) Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve	Bounded by Reference Cycle
7.3.1	5) Partial Loss of Forced Reactor Coolant Flow	Bounded by Reference Cycle
7.3.2	6) Total Loss of Forced Reactor Coolant Flow	Bounded by Reference Cycle
7.4.1	7) Uncontrolled CEA Withdrawal from Subcritical or Low Power Conditions	Bounded by Reference Cycle
7.4.2	8) Uncontrolled CEA Withdrawal at Power	Bounded by Reference Cycle
7.4.3	9) CEA Misoperation Events	Bounded by Reference Cycle
7.6.1	10) Pressurizer Pressure Decrease Events	Bounded by Reference Cycle
7.7.1	11) Asymmetric Steam Generator Events	Bounded by Reference Cycle
B) Limiting Fault Events		
1) Steam System Piping Failures:		
7.1.5.a	a) Pre-Trip Power Excursions	Bounded by Reference Cycle
7.1.5.b	b) Post Trip Analysis	Bounded by Reference Cycle
7.3.3	2) Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft	Bounded by Reference Cycle
7.4.6	3) CEA Ejection	Bounded by Reference Cycle

Table 7.0-5

DBE's Evaluated with Respect to Shutdown Margin Criterion

<u>Section</u>	<u>Event</u>	<u>Results</u>
A) <i>Anticipated Operational Occurrences</i>		
7.1.4	1) Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve	Bounded by Reference Cycle
7.4.4	2) CVCS Malfunction (Inadvertent Boron Dilution)	Bounded by Reference Cycle
7.4.5	3) Startup of an Inactive Reactor Coolant System Pump	Bounded by Reference Cycle
B) <i>Limiting Fault Events</i>		
7.1.5.b	1) Steam System Piping Failures: b) Post Trip Analysis	Bounded by Reference Cycle

Table 7.0-6

Waterford 3 Cycle 6
Core Parameters Input to Safety Analyses

<u>Safety Parameters</u>	<u>Units</u>	<u>Previous Cycle Values (Cycle 5)</u>	<u>Cycle 6 Values</u>
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	3478	3478
Core Inlet Steady State Temperature	'F	542 to 560 (70% power and above) 520 to 560 (below 70% power)	542 to 560 (70% power and above) 520 to 560 (below 70% power)
Steady State RCS Pressure	psia	2000 - 2300	2000 - 2300
Rated Reactor Coolant Flow	GPM	396,000 to 410,000	396,000 to 410,000
Axial Shape Index LCO Band Assumed for All Powers	ASI Units	-.3 to +.3	-.3 to +.3
Maximum CEA Insertion at Full Power	% Insertion of Lead Bank	28	28
	% Insertion of Part-Length	25	25
Maximum Initial Linear Heat Rate	KW/ft	13.4	13.4
Steady State Linear Heat Rate for Fuel Center Line Melt	KW/ft	21.0	21.0
Minimum DNBR CE-1 (SAFDL)		1.26	1.26
Macbeth (Fuel failure limit for post-trip SLB with LOAC)		1.30	1.30

Table 7.0-6 (continued)

Safety Parameters	Units	Previous Cycle Values (Cycle 5)	Cycle 6 Values
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/^{\circ}\text{F}$	-3.3 to +0.5 (below 70% power) -3.3 to 0.0 (70% power and above)	-3.3 to +0.5 (below 70% power) -3.3 to 0.0 (70% power and above)
Shutdown Margin (Value Assumed in Limiting EOC Zero Power SLB)	$\Delta\rho$	-5.15	-5.15
Time for the Average CEA Position to Reach 90% Insertion Following Opening of the Trip Breakers	Seconds	3.0	3.0

7.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

7.1.1 Decrease in Feedwater Temperature

The results are bounded by the Reference Cycle

7.1.2 Increase in Feedwater Flow

The results are bounded by the Reference Cycle

7.1.3 Increased Main Steam Flow

The Increased Main Steam Flow event with a loss of AC Power and the associated coastdown of the Reactor Coolant Pumps is an event for which the number of fuel pins predicted to experience DNB will be calculated. The flatter distribution of fuel rod powers ("pin census") for Cycle 6 was evaluated to quantify the percent of fuel pins experiencing DNB.

Although the percent of fuel pins predicted to experience DNB remains small, the Cycle 6 value for the percent of fuel pins predicted to experience DNB is not bounded by the value reported in the Reference Cycle (Cycle 1) for this event.

Using the method of statistical convolution, the FSAR reported 0.83% of the fuel pins will experience DNB.

7.1.3.1 Identification of Causes

The Increased Main Steam Flow event is conservatively assumed to use up all the thermal margin preserved by COLSS and the other LCO's through the combination of increases in core power and decreases in RCS pressure. The plant is postulated to reach a temporary steady state condition with the peak pin in the core just above the conditions which would result in a CPC Low DNBR Trip.

The perturbation in the secondary system due to the decreased secondary system pressure is then postulated to result in a turbine trip. A Loss of AC power (LOAC) is postulated to immediately accompany the turbine trip. The LOAC is postulated to interrupt power to the RCP's causing a 4 pump coastdown.

As soon as the decrease in flow is detected by the CPC's, the CPC calculation of DNBR will return a value lower than the trip setpoint. The Cycle 6 analysis credits the timing of the CPC calculations of DNBR which yields a reactor trip earlier than waiting for the RCP's to reach the Low Pump Speed setpoint. The Low Pump Speed setpoint was used for reactor trip in the Reference Cycle analysis.

7.1.3.2 Analysis of Effects and Consequences

The Increased Heat Removal part of the transient has not been explicitly modelled for Cycle 6 since it is conservatively assumed that this portion of the event uses up all the thermal margin initially preserved by the LCO's. This assumption is conservative as explicit modelling of the Excess Load portion of the event would demonstrate that thermal margin to the SAFDL exists at the time of LOAC. The analysis performed for Cycle 6 is equivalent to the

analysis of a 4 pump loss of forced reactor coolant flow initiated from conditions at or just above the DNBR SAFDL.

To use up the initial margin, the increased heat removal must occur in the presence of a negative Moderator Temperature Coefficient (MTC). However, a negative MTC has a beneficial effect for a 4 pump loss of forced reactor flow: increased temperature rise across the core before reactor trip would provide negative reactivity feedback with a negative MTC present.

The limiting MTC for the event is then a balance between these two competing effects. A previous parametric study on MTC valid for Cycle 6 determined that a value of $-1.05 \times 10^{-4} \Delta p/^{\circ}F$ was the most limiting MTC value.

The following analytical steps have been used in the Cycle 6 analysis:

1. The 1-D HERMITE code was used to model the flow coastdown and reactor trip portion of the event.
2. The CPC Low DNBR Trip is credited immediately after the beginning of the coastdown of the RCP's. This is done rather than waiting for the pumps to slow to the CPC Low Pump Speed setpoint. Modeled this way, the time from initiation of the coast down until the power is interrupted to the CEA holding coils is reduced from 0.86 seconds to 0.35 second.

If the CPC Low DNBR trip setpoint was not immediately reached, it would mean that additional initial thermal margin must be present at the onset of the coastdown. This additional margin would result in fewer fuel rods experiencing DNB.

3. To avoid unrealistically combining the worst case MTC with the worst case pin census, burnup dependence of MTC was also accounted for. The Cycle was divided between the times in life in which the MTC was more negative than $-2.0 \times 10^{-4} \Delta p/^{\circ}F$ and those in which the MTC was less negative than this value. Thus, a value of $-1.05 \times 10^{-4} \Delta p/^{\circ}F$ was assumed for approximately the first 10,300 MWD/T burnup of Cycle 6 and a value of $-2.0 \times 10^{-4} \Delta p/^{\circ}F$ was assumed for burnups larger than this.

4. The results from HERMITE cases based upon these MTC values were used to evaluate fuel pin DNBR. The percent of fuel pins violating the SAFDL was calculated based on a bounding pin census chosen for each of the two burnup ranges.

Table 7.1.3-1 contains the sequence of events for the Increased Main Steam Flow event with Loss of AC, based on this trip timing. The Excess Load portion of the event begins at time $T = 0$. The initially preserved thermal margin is reduced until some time, ΔT , at which time the core has reached a point just above the DNBR SAFDL as calculated by CPC's. The specifics of ΔT depend upon the rate and severity with which the Increased Main Steam Flow is imposed upon the RCS.

At time ΔT the Loss of Offsite Power is assumed to occur. The CPC's sense the decreased flow and generate a Low DNBR Trip. The power to the holding coils is removed at $\Delta T + 0.35$ seconds. At $\Delta T + 0.95$ seconds the flux in the holding coils has decayed and the CEA's begin to drop into the core.

The transient DNBR reaches a minimum value at $\Delta T + 2.1$ seconds and then begins to increase. The value of this minimum DNBR is 1.073, which is below the Cycle 6 DNBR SAFDL and more adverse than the Reference Cycle result of 1.096. After this minimum value, event recovery proceeds as presented in the FSAR.

7.1.3.3 Results

The maximum number of pins predicted to experience DNB using the method of statistical convolution is less than 3.0% for Cycle 6.

The predicted number of pins with DNBR values which decrease momentarily below the SAFDL could be reduced considerably if other analytical steps were included. Applying power penalties in the CPC neutron power calculations, detailed examination of the coolant flow into the fuel assemblies with fuel pins predicted to fail, and an explicit calculation of the cooldown and actual loss of operating margin are analytical steps which, if further applied, would reduce the reported amount of fuel pins experiencing DNB.

As stated in the Waterford 3 FSAR, even though some fuel pins experience DNB for this event, full damage would not be expected since the maximum clad temperature would be far less than clad temperatures which could lead to clad failure.

7.1.3.4 Conclusions

With the assumption that no additional thermal margin has been set aside at the beginning of the LOAC portion of the event, the maximum number of pins predicted to experience DNB is less than 3% for Cycle 6. This is acceptable since this results in only a small fraction of the fuel experiencing DNB. Further, while fuel failure is not expected, a coolable core geometry would be maintained even if the 3.0% of fuel calculated to experience DNB failed. This event is presented because the predicted maximum number of fuel pins in DNB for Cycle 6, while acceptable, is not bounded by the Waterford 3 Reference Cycle results for this event.

Table 7.1.3-1

Sequence of Events for the Increased Main Steam Flow
in Combination with a Loss of AC Power

Time (sec)	Event	Setpoint or Value
0.0	Malfunction of Control System causes increased steam flow through the Turbine or the Turbine Bypass Valves	----
ΔT	Thermal Margin Initially Preserved by COLSS Depleted, the Hottest Fuel Rod is Just Above the DNBR SAFDL as Calculated by the CPC's. Loss of AC Power is Assumed and the Coast Down of the RCP's Begins.	----
$\Delta T + 0.35$	Low DNBR Trip Generated by the CPC's, Trip Breakers Open	1.26
$\Delta T + 0.95$	CEA's Begin to Drop	----
$\Delta T + 2.10$	Minimum DNBR Occurs	1.076
$\Delta T + 3.429$	Average CEA Position 90% Inserted	----
$\Delta T + 6.0$	Steam Generator Safety Valves Open	1070 PSIA
$\Delta T + 11.0$	Maximum Steam Generator Pressure	1124 PSIA
1,800	Operator takes control of plant	----
14,000	Shutdown Cooling Initiated	----

- 7.1.4 Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve
The results are bounded by the Reference Cycle
- 7.1.5 Steam System Piping Failures:
The results are bounded by the Reference Cycle.
- 7.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM
The results are bounded by the Reference Cycle for all events in this category.
- 7.3 DECREASE IN REACTOR COOLANT FLOWRATE
The results are bounded by the Reference Cycle for all events in this category.
- 7.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES
The results are bounded by the Reference Cycle for all events in this category.
- 7.5 INCREASE IN REACTOR COOLANT SYSTEM INVENTORY
The results are bounded by the Reference Cycle for all events in this category.
- 7.6 DECREASE IN REACTOR COOLANT SYSTEM INVENTORY
The results are bounded by the Reference Cycle for all events in this category.
- 7.7 MISCELLANEOUS (Asymmetric Steam Generator Events)
The results are bounded by the Reference Cycle.

8.0 ECCS ANALYSIS

8.1 LARGE BREAK LOSS OF COOLANT ACCIDENT (LOCA)

8.1.1 Introduction

An Emergency Core Cooling System (ECCS) performance analysis of the limiting break size was performed for Waterford 3 Cycle 6 to demonstrate compliance with 10CFR50.46 (Reference 8-1), the NRC Acceptance Criteria for Light Water Reactors. The analysis justifies an allowable Peak Linear Heat Generation Rate (PLHGR) of 13.4 kW/ft. This PLHGR is equal to the existing Waterford 3 limit. The method of analysis and detailed results which support this value are presented in the following sections.

8.1.2 Methods of Analysis

The ECCS performance analysis for Waterford 3 Cycle 6 consisted of an evaluation of the differences between Cycle 6 and the Reference Cycle analysis. Acceptable ECCS performance was demonstrated for the Reference Cycle in Reference 8-2. The differences between Cycle 6 and the Reference Cycle include: 1) differences in fuel cycle related parameters, and 2) changes due to the debris resistant fuel design which was introduced in Cycle 5 and previously evaluated for its effect on ECCS performance.

The previous evaluation of the debris resistant fuel design concluded that the blowdown and refill/reflood hydraulics calculations performed for the Reference Cycle remain applicable by including a 3°F penalty to the calculated peak clad temperature.

Burnup dependent calculations were performed using the FATES3B version of CE's NRC approved fuel performance code (References 8-4, 8-5, and 8-6) and the STRIKIN-II (Reference 8-7) code to determine the limiting fuel rod conditions for use in the Cycle 6 ECCS performance evaluation.

8.1.3 Results

Table 8-1 presents a comparison of the significant parameters for Cycle 6 and the Reference Cycle. The initial system flow rate, core flow rate, core inlet and outlet temperatures used in the Reference Cycle analysis are the same as those for Cycle 6. The pressure drops across the core along with the number of plugged U-tubes per Steam Generator are also identical. Therefore, it is concluded that the blowdown and refill/reflood hydraulic calculations employed in the Reference Cycle analysis apply to the Cycle 6 analysis.

The limiting fuel rod conditions for the Reference Cycle and Cycle 6 are compared in Table 8-2. The hot rod gas pressure at the limiting burnup for Cycle 6 is not significantly different from the corresponding pressure for the Reference Cycle. The PLHGR in the average channel of the hot assembly increased slightly. The average fuel and centerline temperatures decreased and the gap conductance decreased from the Reference Cycle. A hot rod temperature transient calculation demonstrated that the less favorable radiation heat transfer characteristics are not offset by the lower average fuel and centerline temperatures for Cycle 6. Figure 8-1 shows a plot of PCT versus time for the hot spot location.

The peak clad temperature and maximum local clad oxidation values of 2173°F and 8.4%, respectively, for the Cycle 6 analysis exceeds the corresponding values for the Reference Cycle analysis of 2150°F and 7.93%. The 2173°F Cycle 6 peak clad temperature includes the 3°F PCT penalty due to the debris resistant fuel design. The core wide oxidation value for Cycle 6 is < 0.805%, unchanged from that reported for the Reference Cycle.

8.1.4 Conclusion

The ECCS performance analysis for Waterford 3 Cycle 6 resulted in a peak clad temperature of 2173°F, a maximum local clad oxidation of 8.4%, and a core wide oxidation of < 0.805%. While the PCT and local clad oxidation values exceed the values previously reported for the Reference Cycle, these values are less than the acceptance limits of 2200°F and 17%, respectively. The core wide oxidation value is unchanged from that reported for the Reference Cycle and is less than the acceptance limit of 1%. Therefore, operation at a PLHGR of 13.4 kW/ft and a power level of 3458 MWt (102% of 3390 MWt) will result in acceptable ECCS performance for Cycle 6.

8.2 SMALL BREAK LOSS OF COOLANT ACCIDENT

8.2.1 Introduction

An ECCS performance analysis of the limiting break size was performed for Waterford 3 Cycle 6 to demonstrate compliance with 10CFR50.46 (Reference 8-1), the NRC Acceptance Criteria for Light Water Cooled reactors.

8.2.2 Method of Analysis

The ECCS performance analysis for Waterford 3 Cycle 6 consisted of an evaluation of the differences between Cycle 6 and Cycle 5 and a comparison to the Reference Cycle analysis. Acceptable ECCS performance was demonstrated for the Reference Cycle in Reference 8-3. Acceptable ECCS performance for Cycle 5 was documented in the Cycle 5 Reload Analysis Report. It was determined that all input data for Cycle 6 are bounded by the Cycle 5 data.

8.2.3 Results

The peak cladding temperature, maximum local cladding oxidation, and core wide oxidation values of 1846°F, 1.7%, and 0.28%, respectively, for the Reference Cycle apply conservatively to Cycle 6.

8.2.4 Conclusions

The Cycle 6 Small Break LOCA results remain bounded by the Reference Cycle. Cycle 6 Small Break LOCA results are less severe than those for the Cycle 6 Large Break LOCA, which remains limiting.

Table 8-1

Waterford 3 Cycle 6 ECCS Analysis
Significant System Parameters

Parameter	Reference Cycle	Cycle 6
Reactor Power Level (102% of Nominal), MWt	3458	3458
System Flow Rate (Total), lbm/hr	148.0x10 ⁶	148.0x10 ⁶
Core Flow Rate, lbm/hr	144.0x10 ⁶	144.0x10 ⁶
Core Inlet Temperature, °F	557.5	557.5
Core Outlet Temperature, °F	618.6	618
Number of Plugged U-Tubes per Steam Generator	400	400

Table 8-2

Waterford 3 Cycle 6 ECCS Analysis
Significant Fuel Pin Parameters

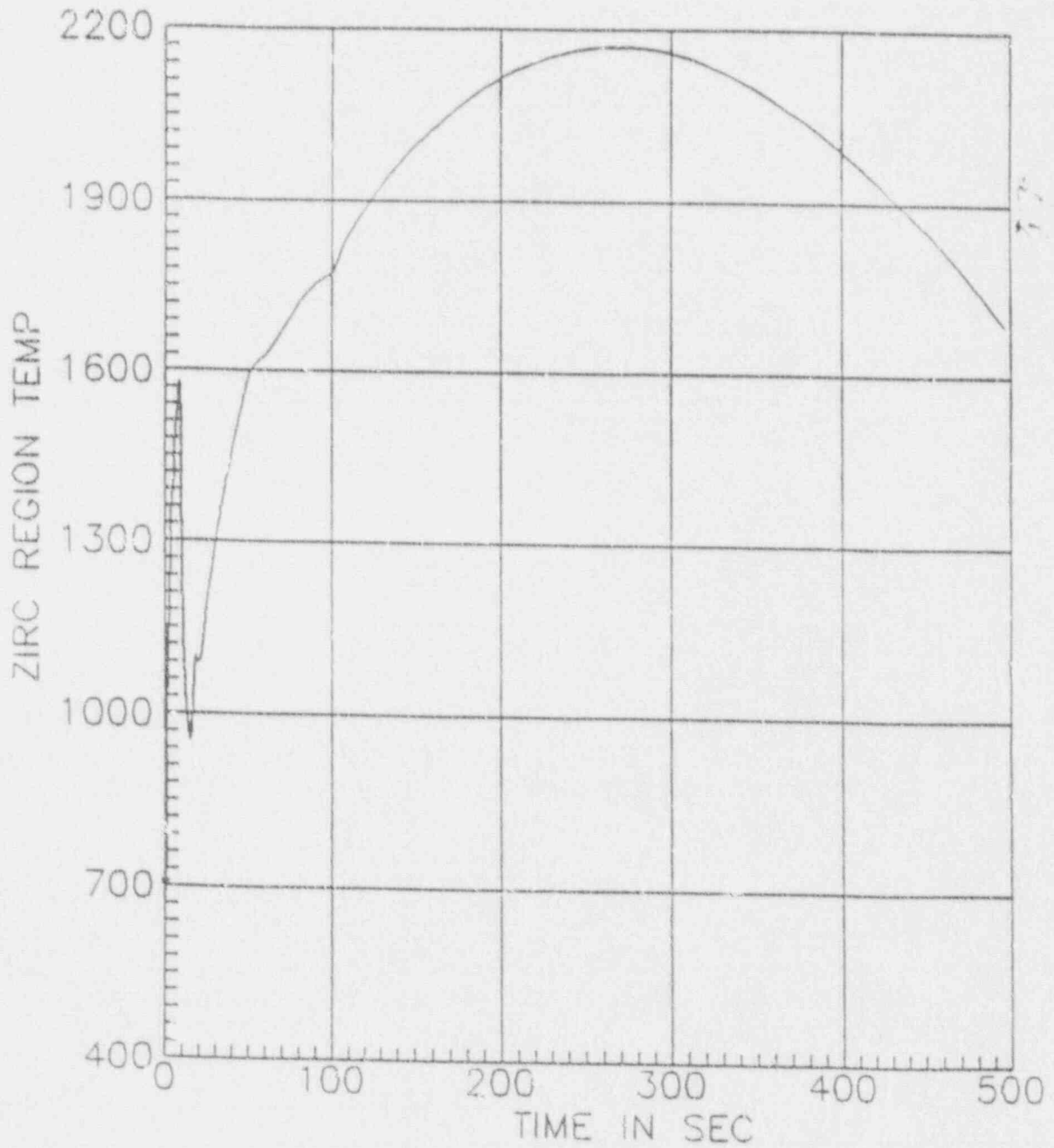
<u>Parameter (1)</u>	<u>Reference Cycle</u>	<u>Cycle 6</u>
Peak Linear Heat Generation Rate, Hot Assembly, Hot Channel, kW/ft	13.4	13.4
Peak Linear Heat Generation Rate, Hot Assembly, Average Channel, kW/ft	12.158	12.409
Fuel Average Temperature at PLHGR, °F	2111.3	2102.1
Fuel Centerline Temperature at PLHGR, °F	3321.6	3290.9
Gap Conductance at PLHGR, BTU/hr-ft ² -°F	1534.0	1513.0
Hot Rod Gas Pressure, psia	1113.3	1114.5
Hot Rod Burnup, MWD/MTU	1000.0	1000.0

(1) The values are at the limiting hot rod burnup as calculated by STRIKIN-II

Figure 8-1

Waterford Unit 3 Cycle 6 ECCS Analysis

PCT (Zirc Region Temp) versus Time for the Hot Spot Location



9.0 REACTOR PROTECTION AND MONITORING SYSTEM

9.1 Introduction

The Core Protection Calculation (CPC) system is designed to provide the low DNBR and high Local Power Density (LPD) trips to (1) ensure that the Specified Acceptable Fuel Design Limits (SAFDL's) on Departure from Nuclear Boiling Ratio (DNBR) and centerline fuel melting (i.e., Local Power Density (LPD) are not exceeded during AOO's, and (2) assist the Engineered Safety Features System in limiting the consequences of certain postulated accidents.

The CPC system, in conjunction with the balance of the Reactor Protection System (RPS), must be capable of providing protection for certain specified Design Basis Events, provided that at the initiation of these occurrences the Nuclear Steam Supply System, its subsystems, components, and parameters are maintained within operating limits and Technical Specification Limiting Conditions for Operation (LCO's).

9.2 CPC Software Modifications

The algorithms associated with the CPC Improvement Program (References 9-1, 9-2, and 9-3) which were implemented in Cycle 2 are applicable to Cycle 6. The values for the Reload Data Block (RDB) constants will be evaluated for Waterford 3 applicability consistent with the Cycle design, performance, and safety analyses. Any necessary change to the RDB constants will be installed in accordance with Reference 9-4.

9.3 Addressable Constants

Certain CPC constants are addressable so that they can be changed as required during operation. Addressable constants include (1) constants related to measurements (e.g., shape annealing matrix, boundary point power correlation coefficients, and adjustments for CEA shadowing and planar radial peaking factors), (2) uncertainty factors to account for processing and measurement uncertainties in DNBR and LPD calculations (BERR0 through BERR4), (3) trip setpoints, and (4) miscellaneous items (e.g., penalty factor multiplier, CEAC penalty factor time delay, pre-trip setpoints, CEAC inoperable flag, calibration constants, etc.). Trip setpoints, uncertainty factors, and other addressable constants will be determined for this cycle consistent with the software and methodology established in the CPC Improvement Program and the Cycle design, performance, and safety analyses.

9.4 Digital Monitoring System (COLSS)

The Core Operating Limit Supervisory System (COLSS), described in Reference 9-5, is a monitoring system that initiates alarms if the LCO's on DNBR, peak linear heat rate, core power, axial shape index, or core azimuthal tilt are exceeded. The COLSS data base and uncertainties will be updated, as required, to reflect the reload core design.

11.0 STARTUP TESTING

The planned startup test program associated with core performance is outlined below. These tests verify that core performance is consistent with the engineering design and safety analysis. Some of the tests also provide the data needed for adjustment of addressable constants in the Core Protection Calculator System (CPC'S) and in COLSS.

11.1 Pre-Critical Test

11.1.1 Control Element Assembly (CEA) Trip Test

Pre-critical CEA drop times are recorded for all full length CEA's at hot, full flow conditions. The drop times will be verified to be within Technical Specification limits.

11.2 Low Power Physics Tests

11.2.1 Initial Criticality

Initial criticality is obtained by fully withdrawing all CEA Groups except Group 6 (which is withdrawn to approximately 75 inches), then diluting the Reactor Coolant System (RCS) until the reactor is critical.

11.2.2 Critical Boron Concentration (CBC)

The CBC is obtained for the All Rods Out (ARO) condition and for a partially rodged configuration. Comparison to the predicted CBC is performed by compensating for the residual CEA worth (from the actual CEA position to the predicted CEA position). The measured CBC's will be verified to be within the equivalent of $\pm 1\% \Delta K/K$ of the design predictions.

11.2.3 Temperature Reactivity Coefficient

The isothermal temperature coefficient (ITC) is measured at the Essentially All Rods Out (EARO) configuration and at a partially rodged configuration. The average coolant temperature is varied and the reactivity feedback associated with the temperature change is measured.

The measured value will be verified to be within $\pm 0.3 \times 10^{-4} \Delta K/K/^{\circ}F$ of the predicted value.

The moderator temperature coefficient (MTC) is calculated by subtracting the predicted value of the fuel temperature coefficient from the ITC. The MTC value is then verified to be within Technical Specification limits.

11.2.4 CEA Reactivity Worth

CEA worths will be measured using the CEA Exchange technique. This technique consists of measuring the worth of a "Reference Group" via standard boration/dilution techniques, then exchanging this group with other groups to measure their worths. All full-length CEA's will be included in the measurement groups. Due to the large differences in relative CEA group worths, two reference groups (one with very high worth and one with medium

worth) may be used. The group to be measured will be exchanged with the appropriate reference group, depending on their predicted worth.

The individual measurement group worths will be verified to be within $\pm 15\%$ or $\pm 0.1\% \Delta K/K$ (whichever is larger) of predicted values. In addition, the total worth of all the measurement groups will be verified to be within $\pm 10\%$ of the predicted total worth.

11.3 Power Ascension Testing

Following completion of the Low Power Physics Test sequence, reactor power will be increased in accordance with normal operating procedures. The power ascension will be monitored by an off-line NSSS performance and data processing computer algorithm. This computer code will be continuously executed in parallel with the power ascension to monitor CPC and COLSS performance relative to the processed plant data against which they are normally calibrated. If necessary, the power ascension will be suspended while necessary data reduction and equipment calibrations are performed. Thus the monitoring algorithm continuously ensures conservative CPC and COLSS operation while optimizing overall efficiency of the test program.

11.3.1 Reactor Coolant Flow

Reactor coolant flow will be measured by calorimetric methods at steady state conditions in accordance with the Technical Specifications. Acceptance criteria will require that the COLSS indicated flow be conservative with respect to the measured flow and that the CPC indicated flow be conservative with respect to the COLSS indicated flow.

11.3.2 Fuel Symmetry Verification

Fixed incore detector data will be examined at lower power to verify that no detectable fuel misloadings exist. Individual instrumented fuel assembly powers will be verified to be within $\pm 10\%$ of the symmetric group average.

11.3.3 Core Power Distribution

Core Power distribution data using fixed incore neutron detectors will be used to further verify proper fuel loading and to verify consistency between the as-built core and the engineering design models. This is accomplished using measurement data from two power plateaus.

Compliance with the acceptance criteria at the intermediate power plateau (between 40 and 70% reactor power) gives reasonable assurance that the power distribution will remain within the design limits while reactor power is increased to 100%.

The final power distribution comparison is performed at full power. Axial and radial power distributions are compared to design predictions as a final verification that the core is operating in a manner consistent with its design within the associated design uncertainties.

The measured results are compared to the predicted values in the following manner for both the intermediate and full power analyses:

- A. The root-mean-square (RMS) error between the measured and predicted radial power distribution for each of the 217 fuel assemblies will be verified to be less than or equal to 0.05.
- B. The RMS error between the measured and predicted axial power distribution for each of the 51 axial nodes will be verified to be less than or equal to 0.05.
- C. The measured values of planar radial peaking factor (F_{xy}), integrated radial peak factor (F_r), core average axial peak (F_z), and the 3-D power peak (F_q) will be verified to be within $\pm 10\%$ of their predicted values.

11.3.4 Shape Annealing Matrix (SAM) and Boundary Point Power Correlation Coefficients (BPPCC'S) Verification

The SAM and BPPCC's are determined from a linear regression analysis of the measured excore detector readings and corresponding core power distribution determined from the incore detector signals. Since these values must be representative for a rodged and unrodged core throughout the cycle, it is desirable to use as wide a range of axial shapes as are available to establish their values. The spectrum of axial shapes encountered during the power ascension has been demonstrated to be adequate for the calculation of the matrix elements.

The incore, excore, and related data are compiled and analyzed throughout the power ascension by the off-line NSSS performance and data processing algorithm. The results of the analysis are used to modify the appropriate CPC constants if necessary.

11.3.5 Radial Peaking Factor (RPF) and CEA Shadowing Factor (CSF) Verification

The RPF's and the CSF's are calculated using fixed incore detector and excore detector data from the following CEA configurations:

- All Rods Out
- Group 6 fully inserted
- Group 6 fully inserted & PLCEA's @ 37.5 inches withdrawn
- PLCEA's @ 37.5 inches withdrawn

Appropriate CPC and/or COLSS constants are modified based on the measured values. The rodged portions of this test may be deleted from the test program if appropriate margin penalties are incorporated into the CPC and COLSS.

11.3.6 Temperature Shadowing Factor Verification

Excore detector response as a function of RCS cold leg temperature during the power ascension will be analyzed by the off-line NSSS performance code to verify the adequacy of the CPC Temperature Shadowing Factor constants.

11.3.7 Reactivity Coefficients at Power

The isothermal temperature coefficient (ITC) is measured at approximately full power by swinging turbine load to alternately increase and decrease core inlet temperature. The swings in core temperature and power are used along with the predicted power coefficient to calculate the ITC. The predicted fuel temperature coefficient is then subtracted from the ITC to obtain the MTC. The measured MTC is then used to verify compliance with the Technical Specifications.

11.4 Procedure if Acceptance Criteria Are Not Met

The results of all tests will be reviewed by the plant's reactor engineering group. If the acceptance criteria of the startup physics test are not met, an evaluation will be performed by the plant's reactor engineering group with assistance from the fuel vendor, as needed. The results of this evaluation will be presented to the Plant Operations Review Committee. Resolution will be required prior to continued power escalation. If an unreviewed safety question is involved, the NRC will be notified.

12.0

REFERENCES

12.1

Section 1.0 References

- (1-1) "Waterford Steam Electric Station, Unit No. 3 Final Safety Analysis Report," Docket No. 50-382.
- (1-2) Letter from K. W. Cook (LP&L) to George W. Knighton (NRC), "Waterford 3 SES, Docket No. 50-382, Reload Analysis Report (RAR)," August 29, 1986.
- (1-3) Letter from K. W. Cook (LP&L) to George W. Knighton (NRC), "Waterford 3 SES, Docket No. 50-382, Reload Cycle 2 Reports," October 1, 1986.
- (1-4) CEN-386-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/Kg for Combustion Engineering 16x16 Pwr Fuel," June 1989.
- (1-5) Letter, A. C. Thadani (NRC) to A. E. Scherer (CE), "Generic Approval of C-E Fuel Performance Code FATES3B (CEN-161(B)-P, Supplement 1-P (TAC NO. M81769)," November 6, 1991.
- (1-6) CEN-356(V)-P-A, Revision 01-P-A, "Modified Statistical Combination of Uncertainties," May, 1988

12.2

Section 2.0 References

None

12.3

Section 3.0 References

None

12.4

Section 4.0 References

- (4-1) D. R. Earles, "WSES-3 Cycle 5 Reload Analysis Report", L-90-040, December 12, 1990.
- (4-2) CEN-386-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/Kg for Combustion Engineering 16x16 PWR Fuel," June 1989.
- (4-3) CEN-161(B)-P Supplement 1-P, "Improvements to Fuel Evaluation Model," Combustion Engineering Inc., August 1986.
- (4-4) CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report," Combustion Engineering Inc., July 1974.

- (4-5) CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," Combustion Engineering Inc., August 1989.
- (4-6) Lester, A. C. Thadani (NRC) to A. E. Scherer (CE), "Generic Approval of C-E Fuel Performance Code FATES3B (CEN-161(B)-P, Supplement 1-P (TAC NO. M81769)," November 6, 1991.

12.5

Section 5.0 References

- (5-1) MSS-NA3-P, "Verification of CECOR Coefficient Methodology for Application Pressurized Water Reactors of the Middle South Utilities System," August 1, 1984.
- (5-2) CENPD-266-P-A, "The ROCS and DIT Computer Codes for Nuclear Design," April, 1983.
- (5-3) CEN-386-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/Kg for Combustion Engineering 16x16 PWR Fuel," June 1989.
- (5-4) FIESTA One Dimensional Two Group Space Time Kinetics Code for Calculating PWR Scram Reactivities," CEN-122, November 1979.
- (5-5) CENPD-275-P-A, "C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers," May 1988.

12.6

Section 6.0 References

- (6-1) CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986.
- (6-2) CENPD-162-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," September 1976.
- (6-3) CENPD-206-P-A, "TORC Code, Verification and Simplified Modeling Methods," June 1981.
- (6-4) CEN-160(S)-P, Rev. 1-P, "CETOP Code Structure and Modeling Methods for San Onofre Nuclear Generating Station Units 2 and 3," September 1981.
- (6-5) CEN-283(S)-P, "Statistical Combination of Uncertainties, Part 1, Combination of System Parameter Uncertainties in Thermal Margin Analysis for San Onofre Nuclear Generating Station Units 2 and 3," June 1984.

- (6-6) CEN-155-(S)-P, "CE-1 Applicability to San Onofre Units 2 and 3 H1D-2 Grids, Response to NRC Questions," March 1981.
- (6-7) CEN-165(S)-P, "Responses to NRC Concerns on Applicability of the CE-1 Correlation to the SONGS Fuel Design," May, 1981.
- (6-8) NUREG-0787, Supplement 1, "Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station, Unit No. 3," Docket No. 50-382, October 1981.
- (6-9) CENPD-225-P-A, "Fuel and Poison Rod Bowing," June 1983.
- (6-10) Robert A. Clark (NRC) to William Cavanaugh III, (AP&L), "Operation of ANO-2 During Cycle 2", July 21, 1981 (Safety Evaluation Report and License Amendment No. 26 for ANO-2).

12.7 Section 7.0 References

- (7-1) "Waterford Steam Electric Station Unit No. 3, Final Safety Analysis Report," Louisiana Power and Light Co., Docket No. 50-382.
- (7-2) Letter from K.W. Cook (LPL) to George W. Knighton (NRC), "Waterford 3 SES, Docket No. 50-382, Reload Cycle 2 Reports," October 1, 1986.
- (7-3) "CESEC, Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981, Enclosure 1-P to LD-82-001, January 6, 1982.
- (7-4) R. V. Macbeth, "An Appraisal of Forced Convection Burnout Data," Proc. Instr. Mech. Engrs., Vol. 180, Pt. 3C, PP 37-50, 1965-1966.
- (7-5) D. H. Lee, "An Experimental Investigation of Forced Convection Burnout in High Pressure Water - Part IV, Large Diameter Tubes at about 1600 psia," A.E.E.W. Report R479, 1966.
- (7-6) CEN-160(S)-P, Rev. 1-P, "CETOP Code Structure and Modeling Methods for San Onofre Nuclear Generating Station Units 2 and 3," September 1981.
- (7-7) CENPD-188-A, "HERMITE Space-Time Kinetics," July 1975.
- (7-8) CENPD-161-P, "TORC Code - A Computer Code for Determining the Thermal Margin of a Reactor Core," July 1975.

- (7-9) CENPD-206-P, "TORC Code Verification and Simplified Modeling Methods," January 1977.
- (7-10) CENPD-183, "Loss of Flow - C-E Methods for Loss of Flow Analysis," July 1975.
- (7-11) D. R. Earles, "WSES-3 Cycle 5 Reload Analysis Report", L-90-040, December 12, 1990.

12.8

Section 8.0 References

- (8-1) Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors, Federal Register, Volume 39, No. 3, January 4, 1974.
- (8-2) Letter from K. W. Cook (LP&L) to NRC, "WSES Unit 3, Docket NO. 50-382, Reanalysis of the Large Break LOCA for Cycle 2", May 18, 1987.
- (8-3) "Waterford Steam Electric Station Unit No. 3, Final Safety Analysis Report," Louisiana Power and Light Co., Docket No. 50-382.
- (8-4) CENPD-139-P-A, "CE Fuel Evaluation Model Topical Report," July 1974.
- (8-5) CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989.

CEN-161(B)-P Supplement 1-P, "Improvements to Fuel Evaluation Model," April 1986.
- (8-6) Letter, A. C. Thadani (NRC) to A. E. Scherer (CE), "Generic Approval of C-E Fuel Performance Code FATES3B (CEN-161(B)-P, Supplement 1-P (TAC NO. M81769)," November 6, 1991.
- (8-7) CENPD-135-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1974.

CENPD-135, Supplement 2-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modifications)," February 1975.

CENPD-135-P, Supplement 4-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976.

CENPD-135-P, Supplement 5-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977.

12.9

Section 9.0 References

- (9-1) CEN-304-P, Rev. 01-P, "Functional Design Requirements for a Control Element Assembly Calculator," May 1986.
- (9-2) CEN-305-P, Rev. 01-P, "Functional Design Requirement for a Core Protection Calculator," May 1986.
- (9-3) CEN-330-P-A, "CPC/CEAC Software Modifications for the CPC Improvement Program Reload Data Block," October 1987.
- (9-4) CEN-323-P-A, "Reload Data Block Constant Installation Guidelines," September 1986.
- (9-5) CEN-312-P, Rev. 01-P, "Overview Description of the Core Operating Limit Supervisory System (COLSS)," November 1986.

12.10

Section 10.0 References

None

12.11

Section 11.0 References

None

APPENDIX A
TO
WATERFORD 3 CYCLE 6 RELOAD ANALYSIS REPORT
EVALUATION OF CHANGE TO NUCLEAR DESIGN METHODS

(1) Description of Change

The original methods and computer codes used to analyze the nuclear design of the core are described in Chapter 4 of the Waterford 3 Safety Analysis Report. A licensee is allowed to make changes to these methods and codes provided that the change does not involve either a Technical Specification change or an Unreviewed Safety Question.

The nuclear design methods and computer codes provide calculated values for the following nuclear design parameters:

- * Reactivity
- * Reactivity Coefficients
- * Control Rod Worths
- * Peaking Factors
- * Power Distribution Related Factors

Several changes have been made to these methods and computer codes to (1) simplify their use, (2) improve their computational efficiency (e.g., the exchange of data between codes), and (3) enhance their calculational accuracy. Of the three types of changes, only the latter, enhancing their calculational accuracy, is most likely to significantly affect the numerical results. Since the results of nuclear design analysis are used as input to the transient safety analysis that considers accidents and malfunction of equipment important to safety, these changes must be evaluated to determine whether or not an unreviewed safety question is created.

The original nuclear design methods and computer codes are described in CE's proprietary Topical Report CENPD-266-P-A, "The ROCS & DIT Computer Codes for Nuclear Design," dated April 1983. This Topical Report was generically reviewed and approved by the NRC. Subsequent to the NRC's approval, changes

were made to the methods and codes that could affect the calculational accuracy of the nuclear design computer codes. These changes are as follows:

- * Implementation of Nodal Expansion Method to ROCS
- * Improved Accounting of Anisotropic Scattering and Higher Order Interface Current Angular Distributions in DIT
- * Use of Assembly Discontinuity Factor² between ROCS and DIT
- * Update of Biases and Uncertainties Applied to Calculated Parameters

A description of each change is provided below. The descriptions provide sufficient detail to perform a safety evaluation.

Nodal Expansion Method

The Nodal Expansion Method (NEM) was added to the ROCS code as an alternative to the original Higher Order Difference (HOD) formulation. The ROCS code provides reactor power distributions and effective neutron multiplication factors. This data is then used to derive control rod worths, depletion, reactivity coefficients and reactivity differentials. Use of the NEM achieves significant reduction in computer running times and also improves agreement with fuel management measurement data².

Although the NEM had not yet been fully integrated into the ROCS code, the use of the NEM was fully described in CE Topical Report CENPD-266 that was approved by the NRC. Specifically, Topical Report CENPD-266 explained that NEM had been incorporated into a version of C-E's coarse-mesh kinetics code, HERMITE. Furthermore, Topical Report CENPD-266 presented numerical comparisons of the NEM and HOD methods for solving the neutron diffusion equations. The results showed that the substitution of NEM for the HOD method in ROCS would not have a significant impact on calculational results and uncertainties.

In recognition of the expected future implementation of the NEM into ROCS, the NRC stated the following in the Safety Evaluation Report (SER) that approved C-E Topical Report CENPD-266:

"We have reviewed the ROCS and DIT computer codes as described in CENPD-266-P and CENPD-266-NP and find them to be acceptable for nuclear core design and safety-related neutronics calculations made by CE in licensing actions for power distributions, control rod worths, depletion, reactivity coefficients and reactivity differential. We also conclude that the ROCS code, including the fine-mesh module MC, is of sufficient accuracy for the generation of coefficient libraries for the in-core instrumentation.

The staff, however, recommends that CE perform further verification when the NEM is incorporated into the ROCS code in order to be assured that equivalent calculational biases and uncertainties are obtained with ROCS-NEM as compared to ROCS-HOD."

Before using ROCS-NEM for nuclear design analysis for Waterford 3, CE performed further verification to confirm that the calculational biases and uncertainties obtained with ROCS-NEM are equivalent to ROCS-HOD. The SER did not require CE to resubmit the ROCS-NEM version of the code to the NRC for approval. It is important to note, however, that the NRC did recommend that the biases and uncertainties obtained when NEM was incorporated into ROCS be equivalent when compared to ROCS-HOD. By equivalent, it is understood that the results between the two methods need not be numerically identical, but rather that the two methods be equal to the degree that the same conservative relationship is maintained between calculated and measured data (i.e., a 95/95 tolerance limit).

CE has confirmed that the ROCS-NEM nuclear core design and safety-related neutronics calculations of power distributions, control rod worths, depletion, reactivity coefficients and reactivity differentials maintain the same conservative relationship between calculated and measured data. In particular, the tolerance limits applied to the calculated results from ROCS-HOD and ROCS-NEM are identically defined as "the value that must be added to the calculated results to assure that 95% of the calculated values will be greater than the "true" value with 95% confidence." Thus, the change which adds NEM to ROCS has been demonstrated to be equivalent to the ROCS-HOD version, which was approved by the NRC.

Anisotropic Scattering and Higher Order Interface Current Angular Distributions

In order to maintain the calculational accuracy in CE Topical Report CENPD-266 when evaluating fuel containing gadolinium as a burnable poison, CE had to improve the way the nuclear design computer code accounted for the effects of anisotropic scattering and higher order interface current angular distributions in the DIT code. The DIT code is a transport theory-based code which performs spectral and spatial calculations in fuel cell and fuel assembly geometries. The DIT calculations provide few group neutron cross sections for use by the ROCS code.

The improved method for accounting for anisotropic scattering and higher order interface current angular distributions was submitted by CE in a generic Topical Report which was reviewed and approved by the NRC³. These approved methods and computer codes are described in CE Topical Report CENPD-275-P Revision 1-P-A, "CE Methodology for Core Designs Containing Gadolinia-Uranium Burnable Absorbers," dated May 1988. Although these changes were motivated by the need to obtain additional calculational accuracy to analyze gadolinium as a burnable poison, the method itself is independent of the burnable absorber used in the core.

Topical Report CENPD-275 was not submitted on a plant specific docket. It was reviewed by the NRC for generic implementation on PWR cores. In recognition of the generic applicability of the improvements made to the DIT code, the NRC stated the following in the SER that approved CE Topical Report CENPD-275:

"We have reviewed the changes made to the DIT and ROCS/MC codes and methodology to accommodate the use of the integral burnable absorber gadolinium in PWR cores. These changes are typical of the types made by the industry for computing gadolinia cores. The

numerical results that were provided show that acceptable agreement has been obtained between detailed calculations and design calculations. We conclude therefore that the changes made to the DIT and ROCS/MC codes and methodology are acceptable."

"We also conclude that the neutronics methods described in the report (DIT, ROCS/MC, and PDQ), as modified, are acceptable for calculating the neutronic characteristics of PWR cores containing up to 8 weight percent gadolinia bearing fuel rods."

It is also important to note that benchmark analysis provided in Topical Report CENPD-275 validated the changes made in the DIT code with B_4C poison that contained no gadolinium. The NRC SER, thus, concluded that the methods described in Topical Report CENPD-275 are acceptable for calculating the neutronic characteristics of PWR cores containing up to 8 weight percent gadolinia bearing fuel rods. This includes the case where the PWR core contains zero weight percent gadolinia by virtue of the fact that many of the assemblies used for benchmarking purposes did not contain any gadolinium bearing fuel rods. Indeed, the NRC also noted in the SER the following:

"The results obtained for the Lead Test Assemblies (LTA) are consistent with those obtained for the non-gadolinium bearing fuel assemblies. The staff concurs with CE's conclusion that these results provide additional validation of the DIT code and methodology."

Assembly Discontinuity Factors

Assembly discontinuity factors (ADF's) are used in the nuclear industry⁴ as a method to eliminate homogenization error in nuclear design analysis where the global heterogeneous solution is known. The use of ADF's improves the internal agreement between the DIT and ROCS codes. The ADF's are derived from the very assembly calculations required by the conventional homogenization methods and, therefore, they do not add any new information to the overall calculational methodology. Thus, the use of the ADF's is expected to improve the accuracy of results obtained from ROCS when compared to DIT. CE has confirmed that the assembly discontinuity factors improve the accuracy of the nuclear design analysis method and computer codes.

Biases and Uncertainties

In view of the above changes that have been made to the methods and nuclear design computer codes, the biases and uncertainties applied to the nuclear design parameters were formally reevaluated by CE. For nuclear design parameters, the bias represents either the average of measured value minus the calculated value, or the average ratio of the difference between the measured value and the calculation value compared to the calculated value. The uncertainty value represents the 95/95 tolerance range for the parameter of interest.

The reevaluation produced revised bias and uncertainty values that are equivalent to those reported in CE Topical Report CENPD-266. By equivalent, it is meant that the results are not numerically identical, but rather that their application preserves the same conservative statistical relationship

between calculated and measured data (i.e., the 95/95 probability / confidence level).

The methods used to generate the new biases and uncertainties are the same as that described in Topical Report CENPD-266, with the exceptions of the method used to determine the bias and uncertainty for the net (N-1) rod worth. In the Topical Report, the bias and uncertainty associated with net (N-1) rod worth were calculated by evaluating individual bank worth measurements.

When CE reevaluated the bias and uncertainty for the (N-1) configuration, CE used the bias and uncertainty associated with the sum of the bank worths (i.e., "total" worth) in lieu of that for individual banks. The use of the total rod worth uncertainty is considered more appropriate than the individual bank worth since the total rod worth configuration is more representative of the higher control rod density of the (N-1) configuration.

This change in the bias and uncertainty used for the (N-1) case remains conservative because actual (N-1) measurements demonstrate that the uncertainty of the (N-1) rod worth is lower than the uncertainty of the total worth. This is expected since the (N-1) configuration is strongly influenced by the reactivity of the unrodded region of the core. Thus, the (N-1) configuration is less sensitive to the precision of the calculated effective control rod cross sections than are either the total or individual bank configurations.

The change in method to calculate the (N-1) rod worth produces equivalent set of bias and uncertainty, wherein the same conservative relationship is maintained between calculated and measured data (i.e., a 95/95 tolerance limit).

(2) Unreviewed Safety Question Determination

The changes to the nuclear design analysis methods and computer codes described above do not require changes to the Technical Specifications. No unreviewed safety question exists.

(3) Safety Evaluation

The determination that the changes to the nuclear design analysis methods and computer codes described above do not create an unreviewed safety question is demonstrated by the following:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report will not be increased by the changes to the nuclear design analysis methods and computer codes described above.

The results of nuclear design analyses are used as inputs to the analysis of accidents or malfunction of equipment important to safety that are evaluated in the safety analysis report. These inputs do not alter the physical characteristics of any component involved in the initiation of an accident or any subsequent equipment malfunction. Thus, there is no increase in the probability of occurrence of an

accident or malfunction of equipment important to safety previously evaluated in the safety analysis report as a result of this change.

The consequences of an accident or malfunction of equipment important to safety evaluated in the safety analysis report is affected by the value of inputs to the transient safety analysis. There is always the potential for the value of the nuclear design parameters to change solely as a result of the new reload fuel core loading pattern. Regardless of the source of a change, an assessment is always made of changes to the nuclear design parameters with respect to their effects on the consequences of accidents and equipment malfunctions previously evaluated in the safety analysis.

If increased consequences are anticipated, compensatory actions are implemented to neutralize any expected increase in consequences. These compensatory actions include, but are not limited to, crediting any existing margins in the analysis or redefining the operating envelope to avoid increase consequences. Thus, the nuclear design parameters are intermediate results and by themselves will not result in a increase in the consequence of accident or malfunction of equipment important to safety evaluated in the safety analysis report.

Therefore, the changes to the nuclear design analysis methods and computer codes described above do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

ii. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report will not be created by the changes to the nuclear design analysis methods and computer codes described above.

As noted above, the results of nuclear design analysis are used as inputs to the transient safety analysis of accidents or malfunction of equipment important to safety that are evaluated in the safety analysis report. These inputs do not alter the physical characteristics of any component involved in the initiation of an accident or any subsequent equipment malfunction. Thus, there is no increase in the possibility of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report as a result of this change.

Thus, the changes to the nuclear design analysis methods and computer codes described above will not create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report.

iii. The margin of safety as defined in the basis for any Technical Specification will not be reduced by the changes to the nuclear design analysis methods and computer codes described above.

Benchmarking of the new nuclear design methods and computer codes has demonstrated that the values of those parameters used in the safety analysis are not significantly changed relative to the values obtained

using the previous methods and computer codes. For any changes in the calculated values that do occur, the reevaluation of the biases and uncertainties ensures that the current margin of safety is maintained. Specifically, use of these revised biases and uncertainties in safety evaluations continues to provide the same statistical assurance that the values of the nuclear parameters used in the safety analysis do not exceed the actual values on at least a 95/95 probability/confidence basis.

The changes to the nuclear design analysis methods and computer codes described above, therefore, do not reduce the margin of safety as defined in the basis for any technical specification.

In conclusion, the changes to the nuclear design analysis methods and computer codes described above do not involve an unreviewed safety question and does not require a change to the Technical Specifications.

References

1. USNRC Letter from Cecil O. Thomas (NRC) to A. E. Scherer (CE), "Acceptance for Referencing of Licensing Topical Report CENPD-266-P, CENPD-266-NP "The ROCS and DIT Computer Codes for Nuclear Design", April 4, 1983
2. R. A. Loretz and S. G. Wagner, "Recent Enhancements to the ROCS/MC Reactor Analysis System," Technical Paper presented at the International Conference on the Physics of Reactor Operation, Design and Computation in Marseille, France, April 23-27, 1990
3. USNRC Letter from Ashok C. Thadani (NRC) to A. E. Scherer (CE), "Acceptance for Referencing of Licensing Topical Report CENPD-275-P, Revision 1-P, "CE Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers," May 16, 1988.
4. K. S. Smith, "Assembly Homogenization Techniques for Light Water Reactor Analysis," Progress in Nuclear Energy, Volume 17, 1986.

APPENDIX B

TO

WATERFORD 3 CYCLE 6 RELOAD ANALYSIS REPORT

DEBRIS RESISTANT FUEL DESIGN

A debris resistant feature has been incorporated into Batch G (the Cycle 5 reload Batch) and follow Batch designs. The basic concept of this feature is to move the fuel and poison column axially upward so that the debris that is caught by the Inconel grid interacts with a lengthened solid end cap instead of the fuel or poison rod cladding.

The differences between the debris resistant design and the previous design are as follows:

- 1) Lengthening the lower end cap on each fuel and poison rod so that the bottom of the clad is within the Inconel spacer grid assembly.
- 2) Removing one fuel pellet from each fuel rod. This reduces the active fuel length from 150. inches to 149.6 inches. Also, the plenum spring was redesigned to minimize its material volume. As a result of these two changes the interior void volume of Batch G and follow Batch fuel rods is within 0.5% of the value for Batch F and previous Batches.
- 3) Removing one burnable poison pellet from each poison rod. This reduces the poison stack length from 136 inches to 135 inches. Also, the upper spacer tube was shortened and the overall rod length was increased by 0.7 inch.
- 4) Shortening the height of the lower end fitting by 0.700 inch. The design of the lower end fitting has been changed to improve the ease of installing the fuel bundle assembly in the core and to standardize the debris-resistant fuel design. These changes to the lower end fitting design will not cause the calculated stress intensities in the various load conditions to exceed the design limits.
- 5) Changing the guide tube to lower end fitting connection to accommodate the 0.700 inch reduction in lower end fitting height.

The design of the Zircaloy and Inconel spacer grid assemblies remains unchanged. The locations of Zircaloy grids with respect to those in previous fuel Batches remains unchanged. The Inconel grid in Batch G and follow Batches is located 0.700 inch lower than it was for previous Batches.

Other minor refinements to the fuel assembly mechanical design made since Cycle 2 include:

- * The perimeter strips for the HID-1 spacer grid assemblies have the location of the corner impact arch lowered with respect to its previous location. This change improves fabrication of the spacer grid assemblies.
- * The poison rod assembly design has been modified by replacing the solid Zircaloy-4 spacers with hollow Zircaloy-4 tubes. By using hollow spacer tubes, the poison rod internal volume is increased, allowing higher burnup poison rods.
- * Starting with Batch F (Cycle 4), poison rod overall length has been increased so that it is the same length as the fuel rod. This design

change will not have any effect on fuel assembly performance since poison rod growth is bounded by fuel rod growth.

The locking discs used in the lower end fitting connection to the guide tube have been redesigned, enabling the use of one design in all CE fuel bundle assemblies. The redesign does not alter any design interface requirements between the fuel bundle assembly and the four alignment pins located on the core support plate, and does not affect the structural integrity of the guide tube connection.