RELOAD ANALYSIS REPORT FOR WATERFORD 3 CYCLE 6

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RELOAD ANALYSIS REPORT FOR WATERFORD 3 CYCLE 6

MAY 1992

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

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 n 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.

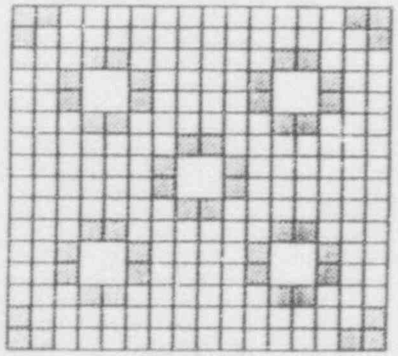
TABLE 3-1

WATERFORD 3
CYCLE 6 CORE LOADING

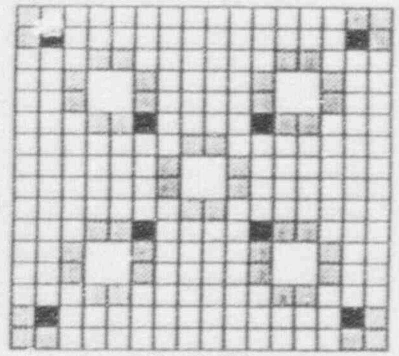
Assembly Designation	Number of Assemblies	Fuel Rods per Assembly	Initial Enrichment (w/o U-235)	Number Poison Rod Assembly	Initial Poison Loading (gm B_{10}/in)	Total Number of Fuel Rods	Total Number of Poison Rods
C1	1	212 12	2.91 2.41	12	0.010	212 12	12
F0	16	184 52	4.05 3.65	0	0	2944 832	0
F1	20	176 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

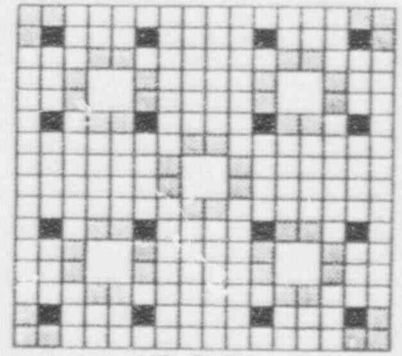
WATERFORD 3 CYCLE 6 FRESH FUEL



H0 FUEL



H1 FUEL



H2 FUEL

FUEL TYPE	No. OF ASSEMBLIES	ENRICHMENT W/O	ENRICHMENT W/O	No. OF SHIMS	GM B-10/IN
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 H[illegible]	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 37300 43900	2 H0 0 14500
				3 F1 35100 40000	4 F1 36900 44300	5 G1 18000 31700	6 H1 0 21400	7 G2 22300 41400
			8 F1 36300 42700	9 H1 0 16300	10 G0 15100 32700	11 H2 0 23100	12 G2 22800 43600	13 H2 0 25300
		14 F2 39400 45600	15 H1 0 16900	16 G2 23200 41100	17 H2 0 24400	18 G1 17300 39500	19 H2 0 25700	20 G2 21800 43600
	21 F1 35100 40000	22 H1 0 16100	23 G2 23200 41100	24 G0 12200 33300	25 G2 21300 42500	26 H2 0 25100	27 F0 34400 52200	28 H2 0 25200
	29 F1 36900 44300	30 G0 15100 32700	31 H2 0 24400	32 G2 21300 42500	33 G0 12200 34000	34 F0 31900 50000	35 H2 0 25200	36 G2 23100 44600
45 F2 37300 43900	37 G1 18000 31700	38 H2 0 23100	39 G1 17300 39500	40 H2 0 25100	41 F0 31900 49900	42 H2 0 25100	43 G2 18600 40700	44 H2 0 25500
54 H0 0 14500	46 H1 0 21400	47 G2 22800 43600	48 H2 0 25700	49 F0 34400 52200	50 H2 0 25200	51 G2 18600 40700	52 G1 17500 38100	53 G2 23100 42200
	55 G2 22200 41400	56 H2 0 25300	57 G2 21800 43600	58 H2 0 25200	59 G2 23100 44600	60 H2 0 25500	61 G2 23100 42200	62 C1 29000 42800

X Y A B

X = Quarter Core Assembly Number

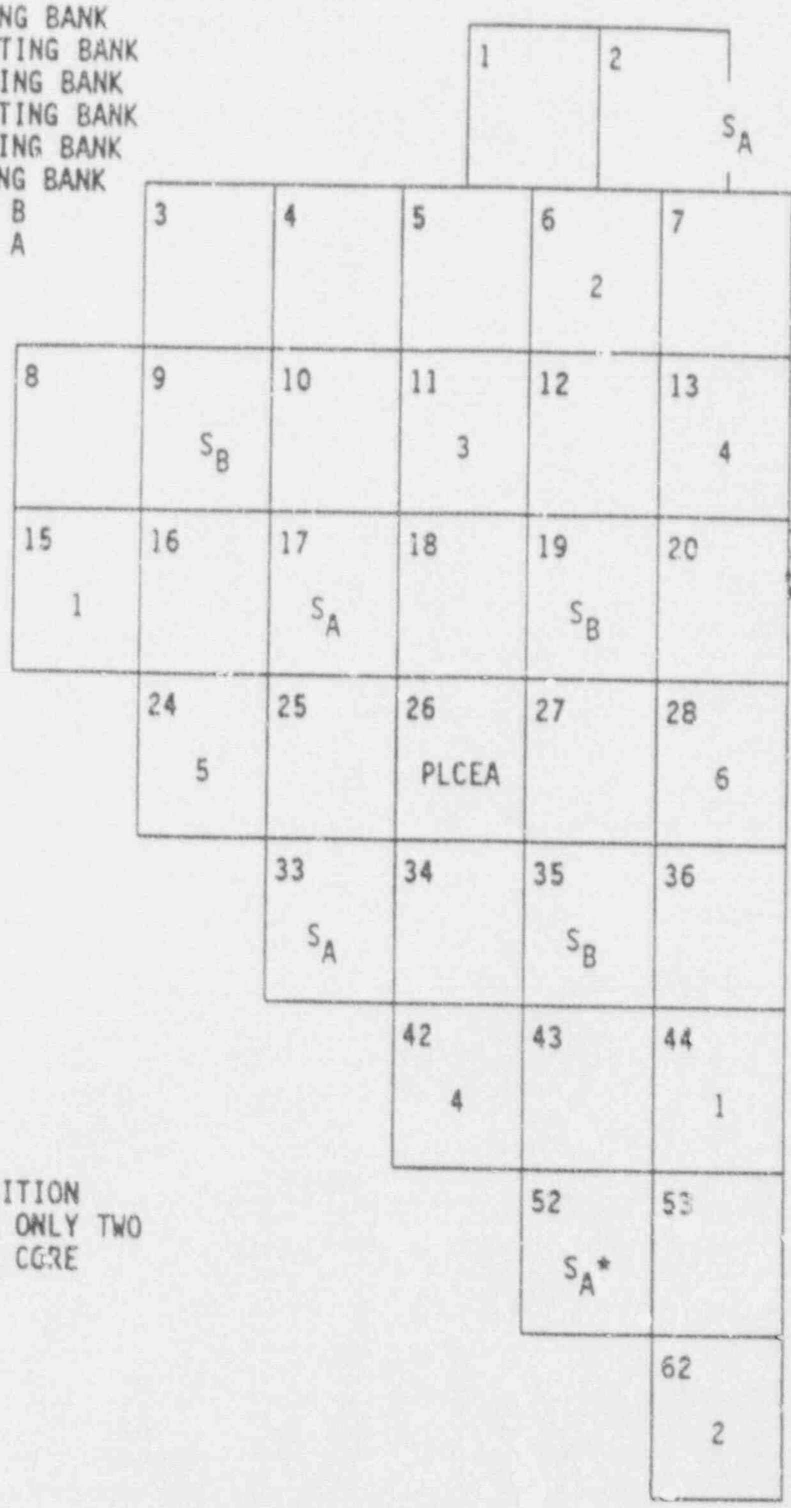
Y = Fuel Batch

A = Assembly Average Burnup (MWD/T), BOC

B = Assembly Average Burnup (MWD/T), EOC

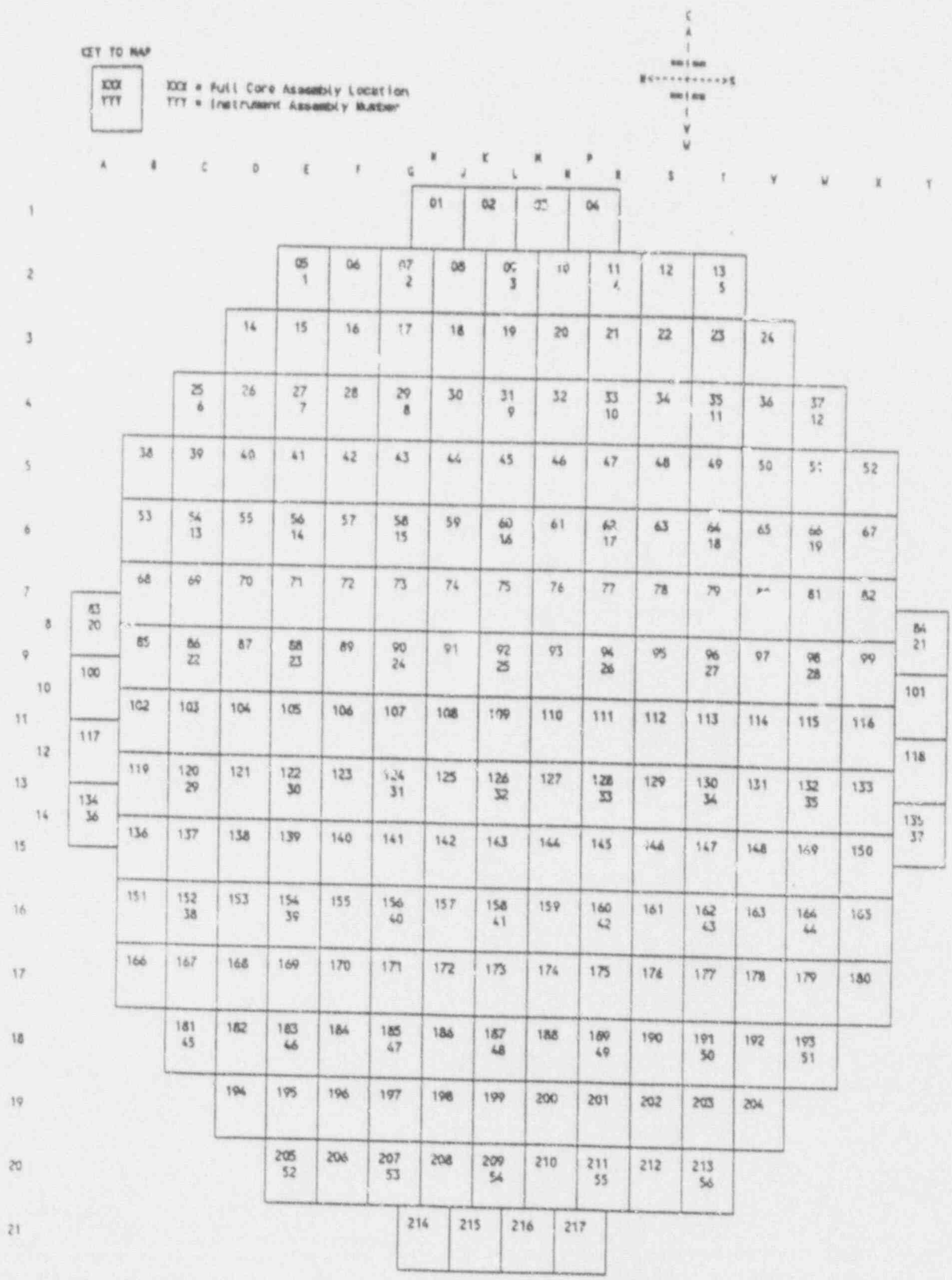
PLCEA - PART LENGTH CEA BANK

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 52 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



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 were explicitly calculated with 3D ROCS (Reference 5-2) while the Reference Cycle values were based on 2D ROCS 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 rodded configurations are given for BOC and EOC in Figures 5-6 through 5-11. The rodded 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 c. 1 be applied without additional NRC review. Additional details are provided in Appendix A t` 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

Dissolved Boron	Units	Reference Cycle	Cycle 6
Dissolved Boron Concentration for Criticality, CEA's Withdrawn, Hot Full Power	PPM	1156	1110
Boron Worth			
Hot Full Power, BOC	PPM/%Δρ	107	130
Hot Full Power, EOC	PPM/%Δρ	85	96
Moderator Temperature Coefficients			
Hot Full Power, Equilibrium Xenon			
Beginning of Cycle	$10^{-4}\Delta\rho/{}^{\circ}F$	-0.1	-0.7
End of Cycle	$10^{-4}\Delta\rho/{}^{\circ}F$	-2.5	-3.1
Doppler Coefficient			
Hot Zero Power, BOC	$10^{-5}\Delta\rho/{}^{\circ}F$	-1.7	-1.7
Hot Full Power, BOC	$10^{-5}\Delta\rho/{}^{\circ}F$	-1.2	-1.2
Hot Full Power, EOC	$10^{-5}\Delta\rho/{}^{\circ}F$	-1.4	-1.4
Total Delayed Neutron Fraction, β_{eff}			
BOC	———	0.0063	0.0062
EOC	———	0.0051	0.0051
Neutron Generation Time, l^*			
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, %Δρ, 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[illegible]
6.	Physics Uncertainty	+ 1.0	+ 0.57
7.	Other Allowances	+ 0.2	+ 0.1
8.	Net Available Scram Worth	- 8.3	- 8.43

TABLE 5-3

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

	Beginning of Cycle		End of Cycle	
Regulating CEA's	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.

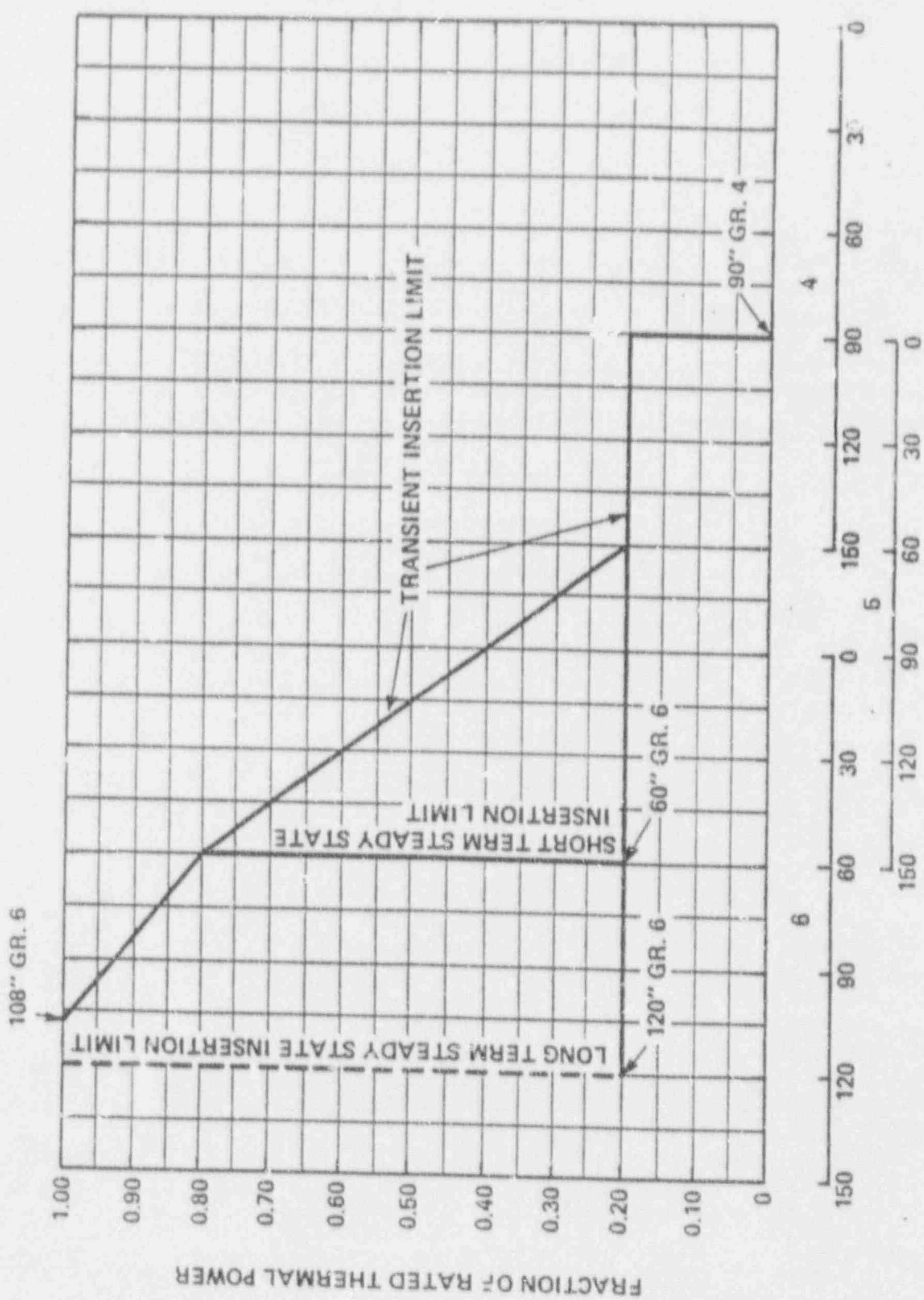
TABLE 5-3

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

	Beginning of Cycle		End of Cycle	
Regulating CEAs	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 5-1
WATERFORD-3
PDIL FOR REGULATING GROUPS



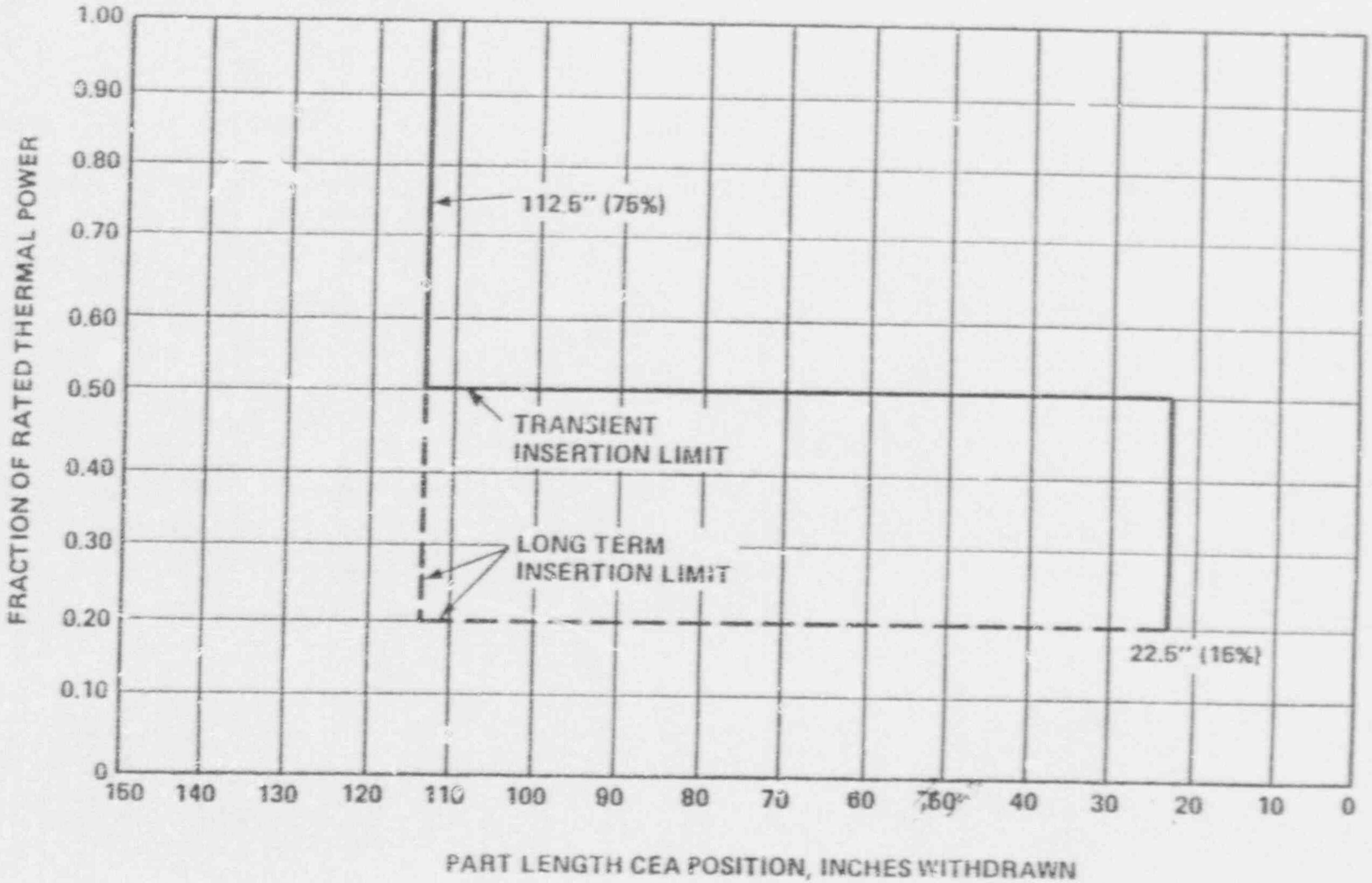


FIGURE 5-2
WATERFORD-3
PART LENGTH CEA INSERTION LIMIT vs THERMAL POWER

Figure 5-3

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

						1 F2 0.348	2 H0 0.841	
				3 F1 0.237	4 F1 0.371	5 G1 0.720	6 H1 1.175	7 G2 1.095
			8 F1 0.324	9 H1 0.852	10 G0 0.953	11 H2 1.180	12 G2 1.127	13 H2 1.303
		14 F2 0.311	15 H1 0.913	16 G2 1.025	17 H2 1.301	18 G1 1.211	19 H2 1.290	20 G2 1.139
	21 F1 0.235	22 H1 0.844	23 G2 1.026	24 G0 1.283	25 G2 1.251	26 H2 1.311	27 F0 0.917	28 H2 1.238
	29 F1 0.370	30 G0 0.951	31 H2 1.302	32 G2 1.250	33 G0 1.301	34 F0 0.993	35 H2 1.278	36 G2 1.138
45 F2 0.348	37 G1 0.719	38 H2 1.180	39 G1 1.210	40 H2 1.307	41 F0 0.989	42 H2 1.321	43 G2 1.224	44 H2 1.340
54 H0 0.841	46 H1 1.175	47 G2 1.127	48 H2 1.289	49 F0 0.916	50 H2 1.276	51 G2 1.224	52 G1 1.191	53 G2 1.104
	55 G2 1.095	56 H2 1.303	57 G2 1.139	58 H2 1.238	59 G2 1.138	60 H2 1.340	61 G2 1.104	62 C1 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 0.351	2 H0 0.771	
				3 F1 0.257	4 F1 0.396	5 G1 0.732	6 H1 1.148	7 G2 1.034
			8 F1 0.335	9 H1 0.857	10 G0 0.941	11 H2 1.237	12 G2 1.127	13 H2 1.369
		14 F2 0.323	15 H1 0.890	16 G2 0.957	17 H2 1.306	18 G1 1.197	19 H2 1.388	20 G2 1.184
	21 F1 0.255	22 H1 0.851	23 G2 0.957	24 G0 1.132	25 G2 1.143	26 H2 1.354	27 F0 0.969	28 H2 1.364
	29 F1 0.395	30 G0 0.939	31 H2 1.306	32 G2 1.142	33 G0 1.174	34 F0 0.978	35 H2 1.366	36 G2 1.170
45 F2 0.351	37 G1 0.732	38 H2 1.237	39 G1 1.197	40 H2 1.351	41 F0 0.976	42 H2 1.361	43 G2 1.207	44 H2 1.386
54 H0 0.771	46 H1 1.148	47 G2 1.126	48 H2 1.388	49 F0 0.968	50 H2 1.365	51 G2 1.207	52 G1 1.119	53 G2 1.035
	55 G2 1.034	56 H2 1.369	57 G2 1.184	58 H2 1.364	59 G2 1.170	60 H2 1.386	61 G2 1.035	62 C1 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

Figure 5-5

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

						1 F2 0.389	2 H0 0.771	
				3 F1 0.321	4 F1 0.465	5 G1 0.782	6 H1 1.138	7 G2 1.001
			8 F1 0.405	9 H1 0.9[illegible]4	10 G0 0.987	11 H2 1.300	12 G2 1.104	13 H2 1.364
		14 F2 0.394	15 H1 0.973	16 G2 0.974	17 H2 1.333	18 G1 1.168	19 H2 1.391	20 G2 1.152
	21 F1 0.320	22 H1 0.949	23 G2 0.973	24 G0 1.069	25 G2 1.075	26 H2 1.344	27 F0 0.975	28 H2 1.370
	29 F1 0.464	30 G0 0.986	31 H2 1.334	32 G2 1.075	33 G0 1.078	34 F0 0.949	35 H2 1.344	36 G2 1.115
45 F2 0.389	37 G1 0.782	38 H2 1.300	39 G1 1.169	40 H2 1.344	41 F0 0.948	42 H2 1.310	43 G2 1.108	44 H2 1.311
54 H0 0.771	46 H1 1.138	47 G2 1.104	48 H2 1.391	49 F0 0.975	50 H2 1.344	51 G2 1.108	52 G1 1.004	53 G2 0.[illegible]39
	55 G2 1.001	56 H2 1.364	57 G2 1.152	58 H2 1.370	59 G2 1.115	60 H2 1.311	61 G2 0.939	62 C1 0.705

Maximum 1-Pin Peak = 1.488 in Assembly 48

X Y Z	
	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 0.355	2 H0 0.837	
				3 F1 0.244	4 F1 0.382	5 G1 0.733	6 H1 1.176	7 G2 1.112
			8 F1 0.333	9 H1 0.858	10 G0 0.966	11 H2 1.177	12 G2 1.142	13 H2 1.302
		14 F2 0.319	15 H1 0.921	16 G2 1.048	17 H2 1.290	18 G1 1.198	19 H2 1.272	20 G2 1.148
	21 F1 0.242	22 H1 0.850	23 G2 1.048	24 G0 1.298	25 G2 1.246	26 H2 1.191	27 F0 0.909	28 H2 1.227
	29 F1 0.381	30 G0 0.963	31 H2 1.290	32 G2 1.245	33 G0 1.295	34 F0 0.989	35 H2 1.266	36 G2 1.154
45 F2 0.355	37 G1 0.733	38 H2 1.176	39 G1 1.197	40 H2 1.187	41 F0 0.985	42 H2 1.318	43 G2 1.246	44 H2 1.352
54 H0 0.838	46 H1 1.176	47 G2 1.142	48 H2 1.271	49 F0 0.908	50 H2 1.265	51 G2 1.245	52 G1 1.229	53 G2 1.148
	55 G2 1.112	56 H2 1.302	57 G2 1.148	58 H2 1.227	59 G2 1.154	60 H2 1.352	61 G2 1.148	62 C1 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 0.364	2 H0 0.856
				3 F1 0.260	4 F1 0.401	5 G1 0.754	6 H1 1.194	7 G2 1.124
			8 F1 0.357	9 H1 0.914	10 G0 1.015	11 H2 1.208	12 G2 1.136	13 H2 1.274
		14 F2 0.342	15 H1 0.987	16 G2 [illegible]7	17 H2 1.362	18 G1 1.233	19 H2 1.209	20 G2 1.014
	21 F1 0.257	22 H1 0.905	23 G2 1.117	24 G0 1.381	25 G2 1.323	26 H2 1.304	27 F0 0.816	28 H2 0.724
	29 F1 0.399	30 G0 1.012	31 H2 1.363	32 G2 1.322	33 G0 1.356	34 F0 1.003	35 H2 1.183	36 G2 1.005
45 F2 0.364	37 G1 0.753	38 H2 1.206	39 G1 1.232	40 H2 1.300	41 F0 0.999	42 H2 1.306	43 G2 1.205	44 H2 1.291
54 H0 0.856	46 H1 1.194	47 G2 1.135	48 H2 1.208	49 F0 0.815	50 H2 1.182	51 G2 1.205	52 G1 1.199	53 G2 1.119
	55 G2 1.124	56 H2 1.274	57 G2 1.014	58 H2 0.724	59 G2 1.005	60 H2 1.291	61 G2 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 0.376	2 H0 0.885
				3 F1 0.264	4 F1 0.408	5 G1 0.770	6 H1 1.226	7 G2 1.156
			8 F1 0.362	9 H1 0.924	10 G0 1.023	11 H2 1.220	12 G2 1.155	13 H2 1.299
		14 F2 0.346	15 H1 0.996	16 G2 1.122	17 H2 1.355	18 G1 1.216	19 H2 1.208	20 G2 1.022
	21 F1 0.262	22 H1 0.915	23 G2 1.121	24 G0 1.374	25 G2 1.296	26 H2 1.187	27 F0 0.800	28 H2 0.722
	29 F1 0.406	30 G0 1.021	31 H2 1.356	32 G2 1.295	33 G0 1.326	34 F0 0.981	35 H2 1.175	36 G2 1.007
45 F2 0.376	37 G1 0.770	38 H2 1.219	39 G1 1.215	40 H2 1.183	41 F0 0.977	42 H2 1.303	43 G2 1.214	44 H2 1.305
54 H0 0.885	46 H1 1.226	47 G2 1.154	48 H2 1.207	49 F0 0.799	50 H2 1.174	51 G2 1.214	52 G1 1.217	53 G2 1.139
	55 G2 1.156	56 H2 1.299	57 G2 1.022	58 H2 0.722	59 G2 1.007	60 H2 1.305	61 G2 1.139	62 C1 0.788

Maximum 1-Pin Peak = 1.602 in Assembly 24

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

Figure 5-9

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

							1 F2 0.402	2 H0 0.797
				3 F1 0.328	4 F1 0.474	5 G1 0.800	6 H1 1.168	7 G2 1.030
			8 F1 0.412	9 H1 0.968	10 G0 0.998	11 H2 1.315	12 G2 1.122	13 H2 1.390
		14 F2 0.401	15 H1 0.986	16 G2 0.981	17 H2 1.326	18 G1 1.152	19 H2 1.389	20 G2 1.160
	21 F1 0.326	22 H1 0.963	23 G2 0.980	24 G0 1.065	25 G2 1.048	26 H2 1.219	27 F0 0.958	28 H2 1.368
	29 F1 0.474	30 G0 0.997	31 H2 1.327	32 G2 1.048	33 G0 1.050	34 F0 0.926	35 H2 1.334	36 G2 1.117
45 F2 0.402	37 G1 0.800	38 H2 1.315	39 G1 1.152	40 H2 1.219	41 F0 0.925	42 H2 1.305	43 G2 1.115	44 H2 1.325
54 H0 0.798	46 H1 1.169	47 G2 1.122	48 H2 1.389	49 F0 0.958	50 H2 1.334	51 G2 1.115	52 G1 1.019	53 G2 0.956
	55 G2 1.030	56 H2 1.390	57 G2 1.160	58 H2 1.368	59 G2 1.117	60 H2 1.325	61 G2 0.956	62 C1 0.721

Maximum 1-Pin Peak = 1.506 in Assembly 56

X Y Z	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 0.417	2 H0 0.825
				3 F1 0.353	4 F1 0.503	5 G1 0.831	6 H1 1.197	7 G2 1.049
			8 F1 0.448	9 H1 1.042	10 G0 1.057	11 H2 1.358	12 G2 1.120	13 H2 1.363
		14 F2 0.435	15 H1 1.068	16 G2 1.057	17 H2 1.412	18 G1 1.188	19 H2 1.312	20 G2 1.009
	21 F1 0.351	22 H1 1.037	23 G2 1.056	24 G0 1.144	25 G2 1.122	26 H2 1.334	27 F0 0.844	28 H2 0.762
	29 F1 0.502	30 G0 1.057	31 H2 1.412	32 G2 1.122	33 G0 1.103	34 F0 0.935	35 H2 1.230	36 G2 0.947
45 F2 0.417	37 G1 0.831	38 H2 1.358	39 G1 1.188	40 H2 1.333	41 F0 0.934	42 H2 1.288	43 G2 1.071	44 H2 1.252
54 H0 0.825	46 H1 1.198	47 G2 1.120	48 H2 1.312	49 F0 0.843	50 H2 1.229	51 G2 1.071	52 G1 0.990	53 G2 0.928
	55 G2 1.049	56 H2 1.363	57 G2 1.009	58 H2 0.762	59 G2 0.947	60 H2 1.252	61 G2 0.928	62 C1 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 0.431	2 H0 0.853
				3 F1 0.361	4 F1 0.514	5 G1 0.851	6 H1 1.229	7 G2 1.079
			8 F1 0.457	9 H1 1.058	10 G0 1.070	11 H2 1.374	12 G2 1.139	13 H2 1.389
		14 F2 0.444	15 H1 1.083	16 G2 1.065	17 H2 1.406	18 G1 1.172	19 H2 1.309	20 G2 1.016
	21 F1 0.359	22 H1 1.052	23 G2 1.064	24 G0 1.141	25 G2 1.095	26 H2 1.209	27 F0 0.824	28 H2 0.757
	29 F1 0.513	30 G0 1.069	31 H2 1.405	32 G2 1.095	33 G0 1.073	34 F0 0.910	35 H2 1.216	36 G2 0.947
45 F2 0.431	37 G1 0.851	38 H2 1.374	39 G1 1.172	40 H2 1.208	41 F0 0.908	42 H2 1.280	43 G2 1.076	44 H2 1.263
54 H0 0.853	46 H1 1.230	47 G2 1.139	48 H2 1.309	49 F0 0.824	50 H2 1.216	51 G2 1.076	52 G1 1.003	53 G2 0.943
	55 G2 1.079	56 H2 1.389	57 G2 1.016	58 H2 0.757	59 G2 0.947	60 H2 1.263	61 G2 0.943	62 C1 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 HID-1 grids as discussed in Reference 6-6, 6-7, and 6-8.

3. Rod bow penalty as discussed in Section 6.. 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

General Characteristics	Units	Reference Cycle (2)	Cycle 6
Total Heat Output (Core only)	MWt	3390	3390
	10^6BTU/hr	11570	11570
Fraction of Heat Generated in Fuel Rod	--	0.975	0.975
Primary System Pressure (Nominal)	psia	2250	2250
Inlet Temperature (Nominal)	°F	553.0	553.0
Total Reactor Coolant Flow	gpm	396,000	396,000
(Minimum Steady State)	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)

General Characteristics	Units	Reference Cycle (2)	Cycle 6
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 Breaks

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

Section		Event	Results
	A)	*Anticipated Operational Occurrences*	
7.1.4	1)	Inadvertent 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
	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
7.2.6	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 RCS Pressure Criterion

Section	Event	Results
	A) Anticipated Operational Occurrences	
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
	B) Limiting Fault Events	
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

Section		Event	Results
	A)	*Anticipated Operational Occurrences*	
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)	*Limiting Fault Events*	
	1)	Steam System Piping Failures:	
7.1.5.b		b) Post Trip Analysis	Bounded by Reference Cycle

Table 7.0-6

Waterford 3 Cycle 6
Core Parameters Input to Safety Analyses

Safety Parameters	Units	Previous Cycle Values (Cycle 5)	Cycle 6 Values
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) Macbeth (Fuel failure limit for post-trip SLB with LOAC)		1.26 1.30	1.26 1.30

Table 7.0-6 (continued)

Safety Parameters	Units	Previous Cycle Values (Cycle 5)	Cycle 6 Values
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/{}^{\circ}F$	-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 -1.05×10^{-4} $\Delta\rho/°F$ 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 $-2.0*10^{-4}$ $\Delta\rho/°F$ and those in which the MTC was less negative than this value. Thus, a value of $-1.05*10^{-4}$ $\Delta\rho/°F$ was assumed for approximately the first 10,300 MWD/T burnup of Cycle 6 and a value of $-2.0*10^{-4}$ $\Delta\rho/°F$ 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 ΔT + 0.35 seconds. At ΔT + 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 ΔT + 2.1 seconds and then begins to increase. The value of this minimum DNBR is 1.07?, 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, fuel 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.	----
ΔT + 0.35	Low DNBR Trip Generated by the CPC's, Trip Breakers Open	1.26
ΔT + 0.95	CEA's Begin to Drop	----
ΔT + 2.10	Minimum DNBR Occurs	1.076
ΔT + 3.429	Average CEA Position 90% Inserted	----
ΔT + 6.0	Steam Generator Safety Valves Open	1070 PSIA
ΔT + 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 ategory.

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	148.0×10^6	148.0×10^6
Core Flow Rate, lbm/hr	144.0×10^6	144.0×10^6
Core Inlet Temperature, °F	557.5	557.5
Core Outlet Temperature, °F	618.6	619
Number of Plugged U-Tubes per Steam Generator	400	400

Table 8-2

Waterford 3 Cycle 6 ECCS Analysis
Significant Fuel Pin Parameters

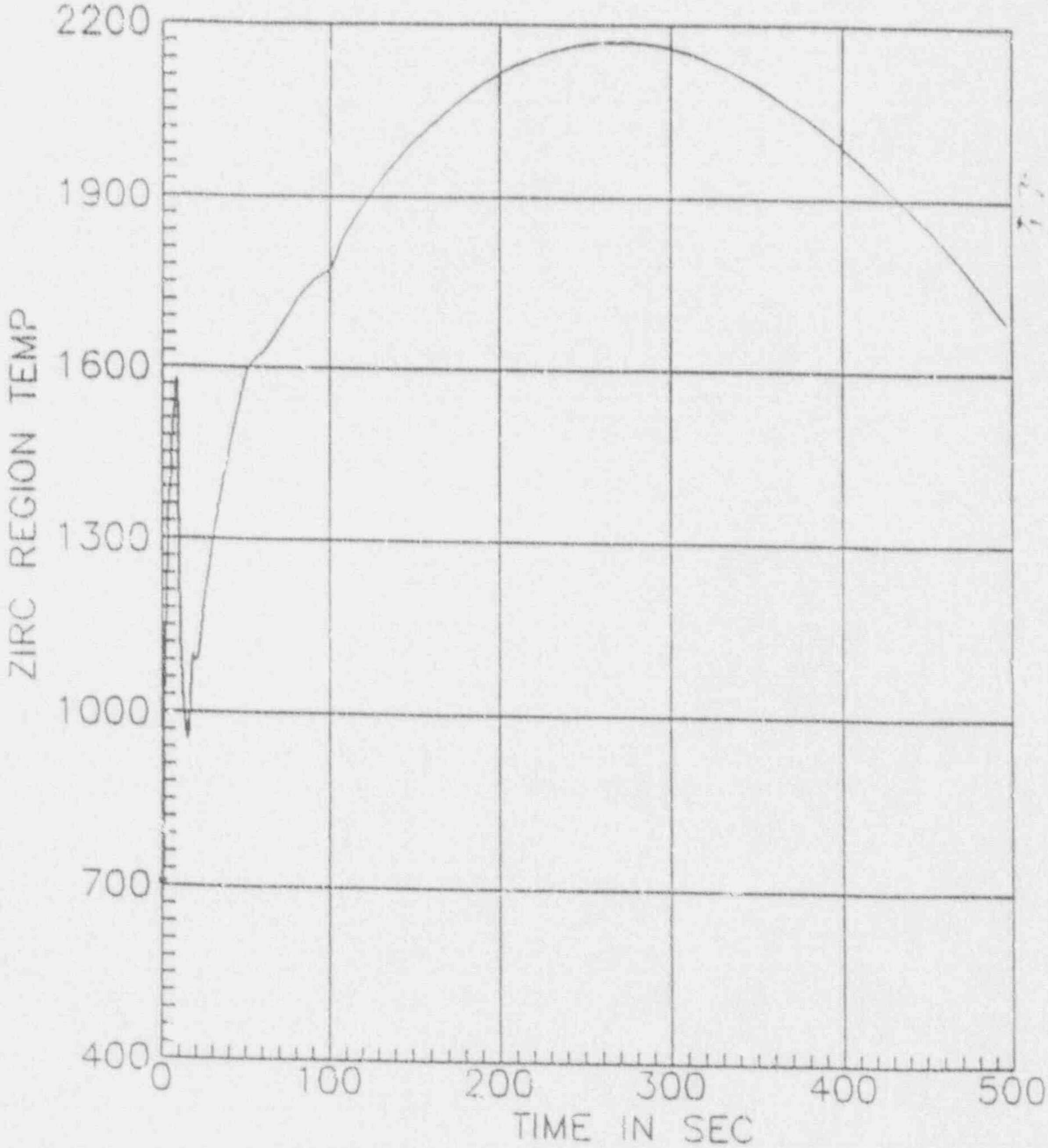
Parameter (1)	Reference Cycle	Cycle 6
Peak Linear Heat Generation Rate, Hot Assembly, Hot Channel, kW/ft	13.4	13.4
Peak Linear Heat Generation Rate, Hot Assembly, Average Channel, kW/ft	12.158	12.409
Fuel Average Temperature at PLHGR, °F	2111.3	2102.1
Fuel Centerline Temperature at PLHGR, °F	3321.6	3290.9
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.

10.0 TECHNICAL SPECIFICATIONS

There are no Technical Specification changes required as a result of the Cycle 6 core design and safety analyses.

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 rodded 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 +/- 1% ΔK/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 rodded 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 +/- 0.3 x 10-4 ΔK/K/°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 groups 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 +/- 15% or +/- 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 +/- 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 +/- 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 +/- 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 rodded and unrodded 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 rodded 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

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(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.

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(4-5) CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," Combustion Engineering Inc., August 1989.

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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 HID-2 Grids, Response to NRC Questions," March 1981.

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(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

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(7-11) D. R. Earles, "WSES-3 Cycle 5 Reload Analysis Report", L-90-040, December 12, 1990.

12.8 Section 8.0 References

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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[1]. 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 Factors 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 data[2].

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 NRC[3]. 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-Urania 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 <u>zero</u> weight percent gadolinia by virtue of the fact that many of the assemblies used for benchmarking purposes did not contain <u>any</u> 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."

<u>Assembly Discontinuity Factors</u>

Assembly discontinuity factors (ADF's) are used in the nuclear industry[4] 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.

<u>Biases and Uncertainties</u>

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:

i. 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 growtn 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.