RELOAD SAFETY EVALUATION

TURKEY POINT PLANT UNIT 4, CYCLE 6

REVISION 1

February, 1979

Edited by

M. D. Beaumont

Work performed under shop order FLFF 14001

APPROVED: M. G. Arlotti

M. G. Arlotti, Manager
Fuel Licensing & Coordination
Nuclear Fuel Division

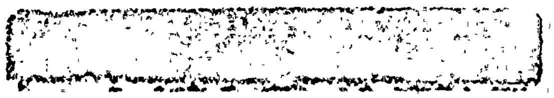

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION AND SUMMARY

Turkey Point Unit 4 is in its fifth cycle of operation. The unit will refuel and be ready for Cycle 6 startup in May, 1979. The Cycle 6 loading pattern shown in Figure 1 contains 40 Region 5, 16 Region 6A, 24 Region 6B, and 12 Region 7 assemblies from Cycle 5; and 65 feed assemblies, 12 at 2.90 w/o, 28 at 3.1 w/o, 24 at 3.35 w/o and 1 at 1.86 w/o. The Cycle 6 fuel inventory is given in Table 1. Two new secondary sources will be used in Cycle 6 along with the two sources from Cycle 5. The location of the sources, depleted burnable poison rods and fresh burnable poison rods is shown in Figure 2.

This report presents an evaluation for Cycle 6 which demonstrates that the core reload will not adversely affect the safety of the plant. It is not the purpose of this report to present a reanalysis of all potential incidents. Those incidents analyzed and reported in the FSAR[1] which could potentially be affected by fuel reloads have been reviewed for the Cycle 6 design described herein. The results of new analyses have been included, and the justification for the applicability of previous results from the remaining analyses is presented. The applicability of the current nuclear design limits was verified for Cycle 6 using the methods described in the Reload Safety Evaluation Methodology WCAP-9273,[2] which includes the PALADON Code[3]. It has been concluded that the Cycle 6 design does not cause the previously acceptable safety limits for any incident to be exceeded. This conclusion is based on the assumption that: (1) Cycle 5 operation is terminated after 4000 $^{+300}_{-300}$ MWD/MTU, (2) Cycle 6 burnup is limited to the end-of-life full power capability* and (3) there is adherence to plant operating limitations discussed later in proposed modifications to the Technical Specifications.

Nominal design parameters for Cycle 6 are 2200 MWt core power, 2250 psia system pressure, 544.8°F core inlet temperature, 255,075 gpm total thermal design flow, and 5.58 kw/ft average linear fuel power density (based on 144" active fuel length).

*Definition: Full-rated power and temperature (approximately 574.2°F vessel T_{avg}), control rods fully withdrawn and zero ppm residual boron.

2.0 REACTOR DESIGN

2.1 Mechanical Design

The mechanical design of Regions 8A, 8B, 8C and 8D fuel assemblies and rods is the same as Region 7. The Region 8 fuel has been designed according to the fuel performance model given in Reference 4. Differences exist in enrichment between Region 7 and Regions 8A, 8C and 8D as shown in Table 1.

Table 1 compares pertinent design parameters of the various fuel regions. The Region 8 fuel is designed and operated so that clad flattening will not occur as predicted by the Westinghouse model (Reference 5). For all fuel regions, the fuel rod internal pressure design basis is revised from not exceeding coolant pressure during normal operation and Condition II accident events to the following. "The internal pressure of the lead rod in the reactor is limited to a value below that which could cause (1) the diametral gap to increase due to outward cladding creep during steady state operation and (2) extensive DNB propagation to occur." Reference 6 shows that the DNB propagation criteria are satisfied.

Westinghouse has had considerable experience with Zircaloy clad fuel. This experience is described in WCAP-8183, "Operational Experience with Westinghouse Cores,"[7] which is updated annually.

2.2 Nuclear Design

The Cycle 6 loading pattern results in a maximum calculated F_Q less than or equal to 2.20 at normal operating conditions. Table 2 provides a comparison of the range of values encompassing the Cycle 6 core kinetics parameters with the current limit based on previously submitted accident analyses. It can be seen from the table that most of the Cycle 6 range of values fall within the current limits. These parameters are evaluated in Section 3.0. Table 3 provides the control rod worths and requirements. The required shutdown margin is based on previously submitted accident analysis.[1] The reactivity defects encompass the values for the Cycle 6 core. The available shutdown margin meets or exceeds the minimum required.

2.3 Thermal and Hydraulic Design

No significant variations in thermal margins will result from the Cycle 6 reload. The core DNB limits at the thermal design flow rate of 255,075 gpm (Reference 8) are applicable for the Cycle 6 design.

3.0 POWER CAPABILITY AND ACCIDENT EVALUATION

3.1 Power Capability

This section reviews the plant power capability considering the consequences of those incidents examined in the FSAR using the previously accepted design bases. It is concluded that the core reload will not adversely affect the ability to safely operate at 100% rated power during Cycle 6. For the overpower transient, the fuel centerline temperature limit of 4700°F can be accommodated with margin in the Cycle 6 core. The time dependent densification model[9] was used for this evaluation. The LOCA limit is met by maintaining F_Q at or below 2.03 with less than or equal to 25% uniform steam generator tube plugging. Assurance that the 2.03 peaking factor is not exceeded will be obtained by performing a surveillance program as discussed in Reference 10 when core power exceeds 92% $\left(\frac{\text{minimum allowable } F_Q \text{ (2.03)}}{\text{maximum calculated } F_Q \text{ for the cycle (2.20)}} \times 100\%\right)$. Alternately, manual APDMS procedures can be used above the 92% turn-on power.

3.2 Accident Evaluation

The effects of the reload on the design basis and postulated incidents analyzed in the FSAR[1] have been examined. In most cases, it was found that the effects can be accommodated within the conservatism of the initial assumptions used in the previous applicable safety analysis. For those incidents which were reanalyzed, it was determined that the applicable design basis limits are not exceeded, and therefore, the conclusions presented in the FSAR are still valid.

A reload can typically affect accident input parameters in three major areas: kinetics characteristics, control rod worths, and core peaking factors. Cycle 6 parameters in each of these areas were examined as discussed below to ascertain whether new accident analyses are required.

Kinetics Parameters

A comparison of the range of values encompassing the Cycle 6 kinetics parameters with the current limits is given in Table 2. Most of the range of values remain within the bounds of the current limits. The moderator temperature coefficient will be zero or negative during normal operation. With the exception of the least negative and most negative Doppler power coefficient, the small changes in core physics parameters have a negligible effect on transient analysis.

For this cycle, the least negative Doppler power coefficient is non-conservative. All transients significantly impacted by this parameter(2) had been reanalyzed in previous cycles using a more conservative value of Doppler than assumed in the FSAR or than calculated in Cycle 6 with the exception of the uncontrolled RCCA bank withdrawal from subcritical transient. This transient was re-analyzed as discussed in Section 3.3. The prompt neutron lifetime has increased to approximately 20μ sec. in the past few cycles.

The most negative Doppler power coefficient for this cycle is less conservative than the value assumed in the FSAR. The only transient impacted by this change is the dropped Bank which is discussed further in Section 3.3. All other transients sensitive to the most negative Doppler power coefficient have been reanalyzed in previous cycles using a more conservative value than assumed in the FSAR or than calculated for Cycle 6.

Control Rod Worths

Changes in control rod worths may affect shutdown margin, differential rod worths, ejected rod worths, and trip reactivity. Table 3 shows that the Cycle 6 shutdown margin requirements are satisfied. As shown in Table 2, the maximum differential rod worth of two RCCA control banks moving together in their highest worth region for Cycle 6 is less than or equal to the current limit.

Core Peaking Factors

Peaking factors following control rod ejection are within the bounds of the current limits. Evaluation of peaking factors for the rod out of position and dropped RCCA incidents shows that DNBR is maintained above 1.30. For the dropped bank incident, the turbine runback setpoint is sufficient to prevent a DNBR less than 1.30.

3.3 Incidents Reanalyzed

Uncontrolled RCCA Bank Withdrawal from Subcritical

This transient was reanalyzed consistent with the methods, assumptions and acceptance criteria presented in the FSAR[1] with the exception of the Doppler coefficient. In addition, a highly conservative reactivity insertion rate (100 pcm/in) and prompt neutron lifetime (26 μ sec) were assumed to provide future design flexibility. The results of the analysis show that all FSAR acceptance criteria are met and therefore the safety conclusion of the FSAR is unchanged.

RCCA Dropped Bank

The re-analysis of this transient is consistent with the FSAR criteria with the exception of the most negative Doppler coefficient. The most negative Doppler coefficient used for this analysis is more conservative than the Doppler power coefficient for this cycle, and therefore provides additional margin for future cycles. This results in less than a 5% increase in peak heat flux from the previously reported analysis. The results of this analysis show that all FSAR acceptance criteria are met and therefore the safety conclusion of the FSAR is unchanged.

4.0 TECHNICAL SPECIFICATIONS

This section references proposed changes to the Technical Specifications. These changes are consistent with plant operation necessary for the design and safety evaluation conclusions stated previously to remain valid.

4.1 Specification 3.2.1, Rod Insertion Limits

Proposed revisions to Technical Specification 3.2.1, "Rod Insertion Limits", Figures 3.2-1 and 3.2-1(a), are given in Reference 11.

While two-loop rod insertion limits have been included, the safety evaluation with these two-loop insertion limits has not been performed since two-loop operation is precluded due to other considerations.

5.0 REFERENCES

1. Final Safety Analysis Report, Turkey Point Units No. 3 and 4.

2. Bordelon, F.M. (et al), "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273, March 1978.

3. Letter NS-TMA-2024 from T. M. Anderson (Westinghouse), to H. R. Denton (US NRC), dated January 17, 1979.

4. Miller, J. V., (Ed), "Improved Analytical Model Used in Westinghouse Fuel Rod Design Computations," WCAP-8785, October 1976.

5. George, R. A., (et al), "Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381 (Non-Proprietary), July 1974.

6. Risher, D. H., (et al), "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8964, June 1977.

7. O'Hara, T. L., Iorii, J. A., "Operational Experience with Westinghouse Cores", WCAP-8183, Revision 7, March, 1978.

8. Amendment Nos. 38 and 31 to Facility Operating License Nos. DPR-31 and DPR-41 for the Turkey Point Nuclear Generating Units 3 and 4, September 22, 1978.

9. Hellman, J. M. (Ed), "Fuel Densification Experimental Results and Model for Reactor Operation", WCAP-8218-P-A, March 1975 (Proprietary) and WCAP-8219-A, March 1975 (Non-Proprietary).

10. Letter from R. E. Uhrig, Florida Power and Light Company, to Victor Stello, D.O.R. Nuclear Regulatory Commission, dated April 10, 1978.

11. Letter FP-FP-424 from R. T. Meyer, Westinghouse NFD Fuel Projects to R. S. Craig, Florida Power & Light Company dated December 14, 1978.

TABLE 1

FUEL ASSEMBLY DESIGN PARAMETERS

TURKEY POINT UNIT 4 - CYCLE 6

Region	5	6A	6B	7	8A	8B	8C	8D
Enrichment (w/o U-235)	3.00	2.90	3.10	3.10	2.90	3.10	3.35	1.86
Density (% Theoretical)*	94.7	94.6	94.7	94.6	95.0	95.0	95.0	94.3
Number of Assemblies	40	16	24	12	12	28	24	1
Approximate Burnup at Beginning of Cycle 6 (MWD/MTU)	23000	15900	11500	4500	0	0	0	0

*All regions except 8A, 8B and 8C are as-built values; Regions 8A, 8B and 8C are the nominal values. However, an average density of 94.5% theoretical was used in the thermal evaluations.

TABLE 2

KINETICS CHARACTERISTICS
TURKEY POINT UNIT 4 - CYCLE 6

Region	Current Limit	Cycle 6
Moderator Temperature Coefficient, $(\Delta\rho/^{o}F) \times 10^{4}$	-3.5 to 0.0	-3.5 to 0.0 ***
Doppler Coefficient $(\Delta\rho/^{o}F) \times 10^{5}$	-1.6 to -1.0	-2.6 to -1.0
Delayed Neutron Fraction βeff (%)	0.44 to 0.72	0.44 to 0.72
Prompt Neutron Lifetime (μ sec)	14 to 18*	20.1
Maximum Differential Rod Worth of Two Banks Moving Together at HZP (pcm/in)**	80*	80

* Current limits will be updated this cycle from 14 to 18 to 26 μ sec, and from 80 to 100 pcm/in per Section 3.3.

** pcm = $10^{-5}\Delta\rho$

*** A positive coefficient does not occur at operating conditions. At operating conditions, the value is zero or negative. At HZP, ARØ the moderator coef. is +0.13 pcm/°F.

TABLE 3

TURKEY POINT 4 - CYCLE 5 AND 6
SHUTDOWN REQUIREMENTS AND MARGINS

	Cycle 5 BOC	Cycle 5 EOC	Cycle 6 BOC	Cycle 6 EOC
Control Rod Worth (%Δρ)				
All Rods Inserted Less Worst Stuck Rod	6.51	6.54	6.21	6.43
(1) Less 10%	5.86	5.89	5.59	5.79
Control Rod Requirements (%Δρ)				
Reactivity Defects (Doppler, T_{avg}, Void, Redistribution)	2.13	2.70	2.27	2.75
Rod Insertion Allowance	0.50	0.50	0.50	0.50
(2) Total Requirements	2.63	3.20	2.77	3.25
Shutdown Margin [(1)-(2)] (%Δρ)	3.23	2.69	2.82	2.54
Required Shutdown Margin (%Δρ)	1.36	1.77	1.00	1.77

Figure 1

Turkey Point Unit 4 Cycle 6 Loading Pattern

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

X	Region Number

Figure 2

Turkey Point Unit 4 Cycle 6

Source and Burnable Poison Locations

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2								8D							
3						8		12D S		8					
4				12D	12		12		12		12	12D			
5				12		12				12		12			
6			8		12		12		12		12		8		
7				12		12				12		12		*	
8		8D	12D										12D	8D	
9		*		12		12				12		12			
10			8		12		12		12		12		8		
11				12		12				12		12			
12				12D	12		12		12		12	12D			
13						8		12D S		8					
14								8D							
15															

D Indicates Depleted BP's
S Indicates Secondary Source
* Indicates New Secondary Source

X Number of Burnable Poison Rods