RELOAD SAFETY EVALUATION
D. C. COOK NUCLEAR PLANT
UNIT 2, CYCLE 3

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

Table 1 compares pertinent design parameters of the various fuel regions. The mechanical design of Region 5 fuel is the same as Region 4 fuel. The Region 5 fuel has been designed according to the fuel performance model in reference 4.

Westinghouse has had considerable experience with Zircaloy clad fuel. This experience is extensively described in WCAP-8183, "Operational Experience with Westinghouse Cores[5]", which is updated periodically.

2.2 NUCLEAR DESIGN

Cycle 3 is required to operate such that the $F_Q(Z)$ XP ECCS analysis limit of $\leq 1.99 \times K(z)$ envelope is not exceeded. Figure 3.2-2 of the technical specification defines the normalized K(z) envelope. Using the present technical specification limits on $F_{xy}(z)$, a conservative load follow analysis has demonstrated that the $1.99 \times K(z)$ F_Q envelope will not be exceeded during Cycle 3 plant operations at or below 91% of rated power. Operation above 91% rated power is allowed by APDMS surveillance as noted in the technical specifications, which assures that the $F_Q(z)$ envelope limit is not exceeded. Table 2 provides a comparison of the Cycle 3 kinetics characteristics with the current limit based on previously submitted accident analysis. It can be seen from the table that all of the Cycle 3 values fall within the current limits, except for the Moderator Density Coefficient. These parameters are evaluated in Section 3. Table 3 provides the end of life control rod worths and requirements at the most limiting condition during the cycle. The required shutdown margin is based on previously submitted accident analysis. The available shutdown margin exceeds the minimum required. The control rod insertion limits remain unchanged from Cycle 2, as given in the technical specifications.

1.0 INTRODUCTION AND SUMMARY

Cook Unit 2 is in its second cycle of operation. The unit is expected to refuel and be ready for Cycle 3 startup in April or May 1981.

This report presents an evaluation for Cycle 3 operation 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 reload have been reviewed for the Cycle 3 design described herein. The applicability of the current nuclear design limits was verified for Cycle 3 using the methods described in reference 2. The results of new analyses have been included, and the justification for the applicability of previous results from the remaining analyses is presented. It has been concluded that the Cycle 3 design does not cause the previously acceptable safety limits for any incident to be exceeded.

The above operational conclusions are based on the assumption that: (1) Cycle 2 operation is terminated between 12200 and 14700 MWD/MTU, (2) Cycle 3 burnup is limited to the end-of-full power capability,* and (3) there is adherence to plant operating limitations given in the current and pending[3] technical specifications and their proposed modifications presented in Section 4.

During the Cycle 2/3 refueling, forty-nine Region 2 and forty-three Region 3 fuel assemblies will be replaced by ninety-two Region 5 assemblies. See Table 1 for the number of fuel assemblies in each region and Figure 1 for the Cycle 3 core loading pattern.

Nominal design parameters for Cycle 3 are 3391 MWt core power, 2280 psia core pressure, nominal core inlet temperature of 543°F, and core average linear power of 5.41 kw/ft.

* Definition: Full rated power and temperature (approximately 574°F T_{AVG}), control rods fully withdrawn, and 10 ppm of residual boron.

Sixty Region 5 fuel assemblies will contain fresh burnable poisons arranged as shown in Figure 1. Two symmetrically located Region 4 fuel assemblies will contain secondary source rods during Cycle 3 (See Figure 1). There will also be two additional secondary source assemblies added in Cycle 3 for irradiation (See Figure 1 for location in Region 4).

2.3 THERMAL AND HYDRAULIC DESIGN

No significant variations in thermal margins result from the Cycle 3 reload. The present core limits, which are documented in reference 1 were found to be applicable for Cycle 3.

3.0 POWER CAPABILITY AND ACCIDENT EVALUATION

3.1 POWER CAPABILITY

The plant power capability is evaluated considering the consequences of those incidents examined in the FSAR and subsequent analyses[(3)(6)] justifying a maximum +5 pcm/^{o}F positive moderator temperature coefficient below 70% power and a 573.8^{o}F T_{avg} for all power levels. A non-positive moderator temperature coefficient is used at 70% power and above. It is concluded that the core reload will not adversely affect the ability to safely operate at 100% of rated power during Cycle 3. For overpower transients, the fuel centerline temperature limit of 4700^{o}F can be accommodated with margin in the Cycle 3 core. The LOCA limit for four loop operation at rated power is met by maintaining F_Q at or below 1.99*, according to the normalized $F_Q(z)$ envelope of Technical Specification Figure 3.2-2. This limit is satisfied by the power control maneuvers allowed by the Technical Specifications, which assure that the Final Acceptance Criteria (FAC) limits are met for a spectrum of small and large LOCA's.

3.2 ACCIDENT EVALUATION

The effects of the reload on the design basis and postulated incidents analyzed in the FSAR for four loop operation 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 core reload can typically affect accident analysis input parameters in three major areas: kinetic characteristics, control rod worths, and core peaking factors. Cycle 3 parameters in each of these three areas were examined as discussed below to ascertain whether new accident analyses were required.

* As stated in Section 2.2, APDMS surveillance to monitor $F_Q(z)$ is required above 91% rated power.

Kinetics Parameters

A comparison of Cycle 3 kinetics parameters with the current limits is presented in Table 2. The parameters which have changed in Table 2 for Cycle 3 reflect operation with a positive moderator coefficient $\leq$ +5 pcm/^{o}F between 0 and 70% power as described in reference 3. Several accidents were reanalyzed (See Section 3.3) to insure that the conclusions presented in the FSAR are still valid. An evaluation of moderator feedback effects for the credible steamline break transient shows that the reactor remains subcritical.

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 3 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 3 is less than the current limit. Cycle 3 ejected rod worths were less than those used for the previous analyses.

Core Peaking Factors

Peaking factor evaluations were performed for the rod out of position and hypothetical steamline break accidents to ensure that the minimum DNB ratio remains above the DNBR design limits. These evaluations were performed utilizing the existing transient statepoint information from the previous reference cycle and peaking factors determined for the reload core design. In each case, it was found that the peaking factor for Cycle 3 resulted in a minimum DNBR which was greater than the design limit DNBR. Consequently, for these accidents no further investigation or analysis was required.

The Cycle 3 control rod ejection peaking factors were within the bounds of the Cycle 2 values. Consequently, no rod ejection reanalysis was required.

3.3 INCIDENTS REANALYZED

The boron dilution at power accident was reanalyzed due to a change in the HFP and HZP critical boron concentrations resulting from the large amount of high enrichment fresh fuel present for Cycle 3. Table 4 gives the pertinent parameters used in the reanalysis.

The boron dilution reanalysis was performed using the same methods described in reference 1 and satisfies all the acceptance criteria specified in reference 1. Therefore, the safety conclusions presented in the FSAR are still valid.

Several accidents were reanalyzed to justify a positive moderator temperature coefficient of +5 pcm/°F between 0 and 70% rated power. These were rod withdrawal from subcritical, rod withdrawal at power, loss-of-flow, locked rotor, loss-of-load and rod ejection. Details of the reanalyses are documented in reference 3. Results verify that the safety conclusions presented in reference 1 remain valid.

4.0 TECHNICAL SPECIFICATION CHANGES

To ensure plant operation consistent with the design and safety evaluation conclusion statements made in this report and to ensure that these conclusions remain valid, several technical specifications will be needed for Cycle 3. These changes are discussed below.

4.1 SPECIFICATION 3/4.2 - POWER DISTRIBUTION LIMITS

(1) In Sections 3.2.1.a.1, 3.2.1.a.2 and 3.2.1.b change 84% of RATED THERMAL POWER to 81%.

(2) Modify technical specification Figure 3.2-1 to the information given in enclosed Figure 2.

(3) In Section 3.2.6 - APPLICABILITY:

Replace: Mode 1 above 94% of Rated Thermal Power
With: Mode 1 above 91% of Rated Thermal Power

(4) In BASES 3/4.2.1, Axial Flux Difference (AFD)

Replace: the 84% values with 81% throughout this section.

4.2 SPECIFICATION 3.1

Replace paragraph 3.1.1.4a with:

a. $\leq 0.5 \times 10^{-4}$ k/k/^{o}F below 70% Rated Thermal Power
$\leq 0.0 \times 10^{-4}$ k/k^{o}F at or above 70% Rated Thermal Power

5.0 REFERENCES

1. D. C. Cook Unit 2 Final Safety Analysis Report, USNRC Docket No. 50-316.

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

3. Letter from American Electric Power to NRC; Subject: Request for D. C. Cook Unit 2 Operation with Positive Moderator Coefficient; AEP-NRC-453, September 22, 1980.

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

5. Skaritka, J. and Iorii, J. A., "Operational Experience with Westinghouse Cores", WCAP-8183 Revision 9, April, 1980.

6. Letter from Westinghouse (G. G. Pennington) to American Electric Power (R. W. Jurgensen); Subject: D. C. Cook Unit 2, Reanalysis for T_{avg} = 573.8^{o}F, Letter No. AEP-79-589, July 1979.

7. Skaritka, J., editor, "Reload Safety Evaluation - D. C. Cook Unit 2 Cycle 2, July 1979.

TABLE 1

FUEL ASSEMBLY DESIGN PARAMETERS
COOK UNIT 2 - CYCLE 3

Region	3	4	5
Enrichment (w/o U235)*	3.09	3.40	3.40
Geometric Density (percent Theoretical)*	94.6	94.5	95.0
Number of Assemblies	21	80	92
Approximate Burnup at Beginning of Cycle 3 (MWD/MTU)	17850	14450	0

* All fuel regions except region five are as-built values: Region five values are nominal. However, an average density of 94.5% theoretical was used for Region 5 evaluations.

TABLE 2

KINETICS CHARACTERISTICS

COOK UNIT 2 - CYCLE 3

	Previous Analysis Value (1)(6)	Cycle 3 Value
Moderator Density Coefficient ($\Delta\rho$/gm/cc)	0 to 0.43	>-.049 to <.43**
Least Negative Doppler - Only Power Coefficient, Zero to Full Power (pcm/ power)*	-10.2 to -6.7	-10.2 to -6.7
Most Negative Doppler - Only Power Coefficient Zero to Full Power (pcm/ power)*	-19.4 to -12.6	-19.4 to -12.6
Delayed Neutron Fraction	.0044 to .0075	.0044 to .0075
Maximum Prompt Neutron Lifetime (μsec)	$\leq$ 26	$\leq$ 26
Maximum Reactivity Withdrawal Rate from Subcritical (pcm/sec)*	$\leq$ 75	$\leq$ 60
Doppler Temperature Coefficient (pcm/°F)*	-1.4 to -2.9	-1.4 to -2.9

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

**The moderator density coefficient is predicted to be less negative than -.049 below 70 percent power and positive at and above 70 percent power conditions.

TABLE 3

SHUTDOWN REQUIREMENTS AND MARGINS
COOK UNIT 2 - CYCLE 2 AND 3

	Four Loop Operation			
	Cycle 2 (7)		Cycle 3	
	BOC	EOC	BOC	EOC
Control Rod Worth (percent $\Delta\rho$)				
All Rods Inserted Less Worst Stuck Rod	5.72	5.70	5.72	5.58
Less 10 percent (1)	5.15	5.13	5.15	5.02
Control Rod Requirements (percent $\Delta\rho$)				
Reactivity Defects (Doppler, Tavg, Void, Redistribution)	2.24	2.72	1.69	2.82
Rod Insertion Allowance (RIA)	0.50	0.50	1.28	0.50
Total Requirements (2)	2.74	3.22	2.97	3.32
Shutdown Margin [(1)-(2)] (percent $\Delta\rho$)	2.41	1.91	2.18	1.70
Required Shutdown Margin (percent $\Delta\rho$)	1.60	1.60	1.60	1.60

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TABLE 4

BORON DILUTION AT POWER PARAMETERS

COOK UNIT 2 CYCLE 3

	Previous Analysis Values	Cycle 3 Values	Used in Analysis
Critical C_B (ppm) - BOC, HFP, No Xe Rods to insertion limits	1500	1894	1894
Critical C_B (ppm) - BOC, HZP, No Xe All rods in less one stuck rod	1150	1309	1309

[illegible]

Figure 1
CORE LOADING PATTERN
D. C. Cook Unit 2 Cycle 3

180°

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					3	5	5	5	5	5	3				
2			3	5	4 5	4	12 5	4	12 5	4	4 5	5	3		
3		3	5	4	4	12 5	4	SS 4	4	12 5	4	4	5	3	
4		5	4	4	16 5	4	16 5	4	16 5	4	16 5	4	4	5	
5	3	4 5	4	16 5	3	12 5	4	16 5	4	12 5	3	16 5	4	4 5	3
6	5	4	12 5	4	12 5	4	12 5	4	12 5	4	12 5	4	12 5	4	5
7	5	12 5	4	16 5	4	12 5	SS 4	4	4	12 5	4	16 5	4	12 5	5
8 (270°)	5	4	4	4	16 5	4	4	3	4	4	16 5	4	4	4	5
9	5	12 5	4	16 5	4	12 5	4	4	SS 4	12 5	4	16 5	4	12 5	5
10	5	4	12 5	4	12 5	4	12 5	4	12 5	4	12 5	4	12 5	4	5
11	3	4 5	4	16 5	3	12 5	4	16 5	4	12 5	3	16 5	4	4 5	3
12		5	4	4	16 5	4	16 5	4	16 5	4	16 5	4	4	5	
13		3	5	4	4	12 5	4	SS 4	4	12 5	4	4	5	3	
14			3	5	4 5	4	12 5	4	12 5	4	4 5	5	3		
15					3	5	5	5	5	5	3				

0°

Key	
SS	- Secondary Source
X	- Number of BP
Y	- Region Number

FIGURE 2

AXIAL FLUX DIFFERENCE LIMITS