Attachment 3 to AEP:NRC:0940H

Unit 1 Cycle 10 Reload
Safety Evaluation Report

RELOAD SAFETY EVALUATION
D.C. COOK NUCLEAR PLANT
UNIT 1, CYCLE 10

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TRFC RR

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This report presents an evaluation for D. C. Cook Unit 1, Cycle 10 design, which demonstrates that the core reload will not adversely affect the safety of the plant. The Cycle 10 evaluation was accomplished utilizing the methodology described in WCAP-9273, "Westinghouse Reload Safety Evaluation Methodology".[1]

The RWST and Accumulator boron concentrations have been increased to 2400 to 2600 ppm with the approval of Technical Specification changes[2].

All of the accidents comprising the licensing bases[3] which could potentially be affected by the fuel reload have been reviewed for the Cycle 10 design described herein. The justification for the applicability of the Cook Unit 1 accident analyses is presented in Sections 3.1 and 3.2.

1.2 GENERAL DESCRIPTION

The Cook Unit 1 Cycle 10 reactor core is comprised of 193 fuel assemblies arranged in the core loading pattern configuration shown in Figure 1. During the Cycle 9/10 refueling, 34 ENC+ fuel assemblies and 46 Westinghouse (W) optimized fuel assemblies (OFAs) will be replaced with W Regions 12A and 12B fresh OFAs. A summary of the Cycle 10 fuel inventory is given in Table 1.

+ENC = Exxon Nuclear Corporation

Consistent with the use of the W Improved Thermal Design Procedure (ITDP) for analyses[4], the core design parameters utilized for Cycle 10 are as follows:

Core Power (MWt)	3250 (3411)**
Core Pressure (psia)	2280+
Vessel Average Temperature (°F)	567.8 (577.1)**
Minimum Measured Flow (gpm)	366,400
Average Linear Power Density (kw/ft) (based on average active fuel stack length of 144 inches)	6.70 (7.03)**

1.3 CONCLUSION

From the evaluation presented in this report, it was concluded that the Cycle 10 design does not result in the safety limits for any incident being exceeded. This conclusion is based on the following:

1. Cycle 9 actual burnup of 16,096 MWD/MTU

2. Cycle 10 burnup is limited to the end-of-full power capability (EOFPC)* plus 1500 MWD/MTU for power coastdown.

3. There is adherence to plant operating limitations as given in the Technical Specifications; no changes are needed for Cycle 10 to operate safely.

4. A steam generator tube plugging level of 3.2% is acceptable provided the Technical Specification requirement on RCS flow (Tech. Spec. Table 3.2-1) can be satisfied.

*Definition - with control rods fully withdrawn and approximately 0 to 10 ppm of residual boron at the Cycle 10 3250 MWt rated power conditions.

** Values used in Thermal Hydraulic Analysis
+ Corresponds to Pressurizer Pressure of 2250 psia

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The new Region 12A and 12B fuel assemblies are W 15x15 OFAs. The mechanical design of the Region 12 fuel assemblies is the same as the Region 11 assemblies, except for the use of 4g plenum springs and radiused (bullet nose) fuel rod bottom end plugs.

The 4g spring provides more gas plenum volume for fission gas release and reduces the potential for pellet chipping. The smaller pellet holddown spring in the fuel rod gas plenum satisfies a change in the nonoperational 6g loading design criterion to "4g axial and 6g lateral loading." Westinghouse has incorporated this criterion change, and the justification of no unreviewed safety question was transmitted to the NRC via Reference 5.

The fuel rod bottom end plug was changed only from a chamfered end to a radiused end to improve rod loading and reduce the potential of grid damage during rod loading. This minor design change satisfies all applicable FSAR design criteria. Also, the impact of this change has been evaluated with respect to the licensing basis accident events (See Section 3.2).

Table 1 presents a comparison of pertinent design parameters of the various fuel regions. The Region 12A and 12B fuel have been designed utilizing the W fuel performance model[6] and the W clad flattening model[7]. The W fuel is designed and operated so that clad flattening will not occur for its planned residence time in the reactor. The fuel rod internal pressure design basis [8] is satisfied for all fuel regions.

Westinghouse's experience with Zircaloy clad fuel and OFAs is described in Reference 9. This report is updated annually.

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2.2 NUCLEAR DESIGN

For the Cycle 10 nuclear design the Westinghouse Advanced Nodal Code (ANC),Reference 10, was introduced to perform core neutronics analyses. This supplemented the standard reload methodology design codes given in Reference 1. ANC incorporates several improvements to the PALADON code used in previous core designs. A favorable Safety Evaluation Report (SER), Reference 11, was received from the NRC approving the use of the ANC for nuclear design analyses.

The nuclear design of the Cycle 10 core used W codes approved by the NRC and the standard calculational methods described in the W Reload Safety Evaluation Methodology.[1] The Cycle 10 core loading satisfies the approved technical specification [F_Q (Z) x P] LOCA envelope limit of $\leq$ 2.10 (Figure 2 and Section 3.1).

Table 2 provides a comparison of the Cycle 10 kinetics characteristics with the evaluation limits based on the accident analyses[2,3]. It can be seen from the Table 2 parameters that all of the Cycle 10 values, except the Doppler temperature coefficient, fall within the evaluation limits. (These parameters are evaluated in Section 3.0.) Table 3 provides the beginning and end-of-life control rod worths and requirements at the most limiting condition during the cycle; the available shutdown margin exceeds the minimum required. The control rod insertion Technical Specification limits assure that peaking factors are not exceeded during anticipated power control maneuvers.

Region 12A and 12B fuel assemblies contain a total of 544 new WABA rods (Figure 1). These rods are required for moderator temperature coefficient and power peaking control. Four previously irradiated secondary source rods will be located in two Region 11A assemblies and two Region 12B assemblies (see Figure 1).

2.3 THERMAL AND HYDRAULIC DESIGN

No variation of thermal margins resulted from the all W 15x15 OFA core for the Cycle 10 reload. The present core safety limits in the technical specifications are conservative for the Cycle 10 reload core. Sufficient DNB margin exists for all Condition I and II events and the steamline break transient to meet the design criteria[3,12] for the Cycle 10 reload core.

The table below shows the relationships which exist between the correlation limit DNBR, design limit DNBR, and the safety analysis limit DNBR values used for this design, using the W Improved Thermal Design Procedure (ITDP)[4].

	Typical	Thimble
Correlation Limit	1.17	1.17
Design Limit	1.32	1.31
Safety Analysis Limit	1.69	1.69

For events where conditions fall outside the range of applicability of the WRB-1 correlation (and ITDP), the W-3 correlation is used with the following limits:

Correlation Limit	1.30 for core pressures > 1000
	1.45 for pressures 500 to 1000 psia

The transition core penalty (5% DNBR) applied on the OFA for the Cycle 9 W/ENC mixed fueled core is no longer applicable, since the Cycle 10 core contains only W OFAs.

3.0 POWER CAPABILITY AND ACCIDENT EVALUATION

3.1 POWER CAPABILITY

The plant power capability for Cycle 10 is evaluated by considering the consequences of those FSAR incidents which appear as the licensing basis accident analysis[3,13]. It is concluded that the core reload will not adversely affect the ability to safely operate at the current 3250 MWt rated power during Cycle 10. For overpower transients, the fuel centerline temperature limit of 4700°F can be accommodated with margin in the Cycle 10 core. The time dependent densification model[14] was used for fuel temperature evaluations. The LOCA limit at 3250 MWt is met by maintaining [F_Q(z)xP] at or below [2.10* xK(Z)], according to the normalized F_Q envelope shown in Figure 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 postulated licensing bases incidents[3] were examined. Supporting safety analyses[13], submitted with the Technical Specification changes, justify a number of plant changes (See Section 1.1). Also, an evaluation was performed for the fuel rod bottom end plug change identified in Section 2.1. In all cases it was found that the effects can be accommodated within the conservatism of the initial assumptions used in the applicable safety analysis[3, 13].

An evaluation has demonstrated that the Cycle 10 reactor satisfies the subcritical requirement following a postulated large break LOCA.

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*The 2.10 maximum F_Q value allows a maximum of 5 percent steam generator tube plugging.

In addition, the impact of a 3.2% steam generator tube plugging level has been examined. The conclusions of the current safety analyses remain valid provided the Tech. Spec. requirement on RCS flow (Tech. Spec. Table 3.2-1) are satisfied.

A core reload can affect accident analysis input parameters in the following areas: core kinetic characteristics, control rod worths, and core peaking factors. Cycle 10 parameters in each of these areas were examined, as discussed below, to ascertain whether new accident analyses were required.

3.2.1 Kinetics Parameters

A comparison of Cycle 10 core physics parameters with current evaluation limits is given in Table 2. All the kinetics values, except the Doppler temperature coefficient (see Table 2), remain within the bounds of the analysis limits.

The least negative Doppler temperature coefficient is -1.1 pcm/°F compared to a limit of -1.4 pcm/°F. This coefficient is used in conjunction with the Doppler power coefficient to provide a correction to the power coefficient for fuel temperature changes in transients where the core water temperature changes. This difference, however, has been determined to result in a negligible effect on the accident analyses.

3.2.2 Control Rod Worths

Changes in control rod worths may affect differential rod worths, shutdown margin, ejected worths, and trip reactivity. Table 2 shows that the maximum differential rod worth of two RCCA control banks moving together in their highest worth region for Cycle 10 is less than or equal to the analysis limit. Table 3 shows that the Cycle 10 shutdown margin requirements are satisfied. Cycle 10 ejected rod worths are within the bounds of the analysis limits.

3.2.3 Core Peaking Factors

Evaluation of peaking factors for the rod out of position and dropped RCCA incidents shows that the DNBR is maintained above the appropriate safety analysis limit DNBR value listed in Section 2.3. The peaking factors for the dropped RCCA incidents were evaluated, based on the approved new dropped rod methodology(15).

The hypothetical steamline break transients were evaluated for Cycle 10. This evaluation showed that the Cycle 10 peaking factors are within the bounds of the previous analysis, and DNBR limits (See Section 2.3) are satisfied.

4.0 REFERENCES

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

2. Letter from D. L. Wigginton (USNRC) to John Dolan (I&MECo), dated June 10, 1987, Subject: NRC Approval of Amendment Nos. 111 and 94 to Cook Unit 1 and 2 Technical Specifications.

3. Updated Final Safety Analysis Report - D. C. Cook Unit Number 1, Docket Number 50-315, updated through 1986.

4. Chelemer, H. et al., "Improved Thermal Design Procedure," WCAP-8567, July 1975.

5. Letter from E. P. Rahe, Jr. (Westinghouse) to L. E. Phillips (NRC) dated April 12, 1984, NS-EPR-2893, Subject: Fuel Handling Load Criteria (6g vs. 4g)

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

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

8. Risher, D. H. et.al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8964-A, August, 1978.

9. Skaritka, J., (Ed.), "Operational Experience with Westinghouse Cores (through December 31, 1986)," WCAP-8183, Revision 15, June 1987.

10. Liu, Y. S., et. al., "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10966-A, dated September 1986.

11. NRC Letter from C. Berlinger (NRC) to E. P. Rahe, (Westinghouse) "Acceptance for Referencing of Licensing Topical Report WCAP-10965-P, dated June 23, 1986.

12. Westinghouse Letter dated March 25, 1986, NS-NRC-86-3116, "Westinghouse Response to Additional Request on WCAP-9226-P/WCAP-9227-N-P, "Reactor Core Response to Excessive Secondary Steam Release," (Non-Proprietary).

13. Letter from M. P. Alexich (Indiana and Michigan Electric Co.) to H. R. Denton (USNRC), letter AEP:NRC:091GW dated March 26, 1987, Subject: Proposed Technical Specification for Unit 1 Cycle 10 Reload and Related Unit 2 Proposals, Docket Nos. 50-315 and 50-316.

14. Hellman, J. M. (Ed), "Fuel Densification Experimental Results and Model for Reactor Operation," WCAP-8319-A, March 1975.

15. Morita, T., et. al., "Dropped Rod Methodology for Negative Flux Rate Trip Plants," WCAP-10298-A, June 1983.

TABLE 1

FUEL ASSEMBLY DESIGN PARAMETERS
D. C. COOK UNIT 1 - CYCLE 10

Region	10A	10B	11A	11B	12A	12B
Enrichment (w/o of U 235)*	3.297	3.598	3.404	3.600	3.298	3.600
Density (percent theoretical)*	95.143	95.295	94.995	95.042	95.286	94.911
Number of Assemblies	12	22	47	32	48	32
Burnup at Beginning of Cycle 10 (MWD/MTU)**	28798	27073	19935	17103	0	0
Fuel Stack Height (inches, cold)	144	144	144	144	144	144

* All values are as-built.

** Assumes a Cycle 9 actual core average burnup of 16,096 MWD/MTU

TABLE 2

KINETICS CHARACTERISTICS
D.C. COOK UNIT 1 - CYCLE 10

	Reference Analysis Values (3)(13)	Cycle 10
Moderator Temperature Coefficient, (PCM/°F)*	+5.0 (<70% RTP) +5.0 linear ramp to 0.0. from 70 to 100% RTP	+5.0 (<70% RTP) 0.0 (≥70% RTP)
Doppler Coefficient (PCM/°F)	-2.9 to -1.4	-2.9 to -1.1
Delayed Neutron Fraction β_{eff} (percent)	0.44 to 0.75	0.44 to 0.75
Maximum Prompt Neutron Lifetime (μ sec)	26	≤ 26
Maximum Differential Rod Worth of Two Banks Moving Together at HZP (PCM/sec)*	75	≤ 75

*1 PCM = $1.0 \times 10^{-5}\ \Delta\rho$

TABLE 3

SHUTDOWN REQUIREMENTS AND MARGINS
D. C. COOK UNIT 1 - CYCLE 10

	Cycle 10 BOC	Cycle 10 EOC
Control Rod Worth (percent $\Delta\rho$)		
All Rods Inserted Less Worst Stuck Rod	6.381	6.959
(A) Less 10%	5.743	6.263
Control Rod Requirements (percent $\Delta\rho$)		
Reactivity Defects (Doppler, T_{avg}, Void, Redistribution)	1.774	2.848
Rod Insertion Allowance	0.50	0.50
(B) Total Requirements	2.274	3.348
Shutdown Margin [(A)-(B)] (percent $\Delta\rho$)	3.469	2.915
Required Shutdown Margin (percent $\Delta\rho$)	1.6	1.6

FIGURE 1

D. C. COOK UNIT 1 CYCLE 10 REFERENCE LOADING PATTERN

180°

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					10B	12A	12B	10B	12B	12A	10B				
2			11A	11B	12B	11B	12B 12	11A	12B 12	11B	12B	11B	11A		
3		10A	10B	12B 4	11A	12A 12	11B	11A SS	11B	12A 12	11A	12B 4	10B	10A	
4		11B	12B 4	11B	12A 12	11A	12A 8	11A	12A 8	11A	12A 12	11B	12B 4	11B	
5	10A	12B	11A	12A 12	10A	12A 12	10B	11A	10B	12A 12	10A	12A 12	11A	12B SS	10A
6	12A	11B	12A 12	11A	12A 12	10B	12A 8	11A	12A 8	10B	12A 12	11A	12A 12	11B	12A
7	12B	12B 12	11B	12A 8	10B	12A 8	11B	11A	11B	12A 8	10B	12A 8	11B	12B 12	12B
8 (90°)	11A	11A	11A	11A	11A	11A	11A	11A	11A	11A	11A	11A	11A	11A	11A
9	12B	12B 12	11B	12A 8	10B	12A 8	11B	11A	11B	12A 8	10B	12A 8	11B	12B 12	12B
10	12A	11B	12A 12	11A	12A 12	10B	12A 8	11A	12A 8	10B	12A 12	11A	12A 12	11B	12A
11	10A	12B SS	11A	12A 12	10A	12A 12	10B	11A	10B	12A 12	10A	12A 12	11A	12B	10A
12		11B	12B 4	11B	12A 12	11A	12A 8	11A	12A 8	11A	12A 12	11B	12B 4	11B	
13		10A	10B	12B 4	11A	12A 12	11B	11A SS	11B	12A 12	11A	12B 4	10B	10A	
14			11A	11B	12B	11B	12B 12	11A	12B 12	11B	12B	11B	11A		
15					10B	12A	12B	10B	12B	12A	10B				

0°

X — Region Number
Y/SS — Number of Fresh Burnable Absorbers/ Location of Secondary Source Rods

FIGURE 2

HEAT FLUX HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE FOUR LOOP OPERATION, FQ ECCS LIMIT = 2.10