ENCLOSURE

RELOAD SAFETY EVALUATION

SEQUOYAH NUCLEAR PLANT
UNIT 1, CYCLE 3

ZAR

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

Sequoyah Unit 1 is in its second cycle of operation. The unit is expected to refuel and be ready for Cycle 3 startup in March 1984.

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, Reference 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. It has been concluded that the Cycle 3 design does not cause the previously acceptable safety limits for any incident to be exceeded.

1.2 GENERAL DESCRIPTION

The above operational conclusions are based on the assumption that: (1) Cycle 2 nominal burnup is 11,166 MWD/MTU with a +600 MWD/MTU coastdown and a negative burnup window of 500 MWD/MTU; (2) Cycle 3 burnup is limited to 14,750 MWD/MTU including coastdown; and (3) there is adherence to plant operating limitations given in the technical specifications. The impact of this reload has been evaluated and found to be within the safety limits as described in this RSE for Cycle 3.

During the Cycle 2/3 refueling, five Region 1 and sixty-seven Region 2 fuel assemblies will be replaced by seventy-two Region 5 assemblies. See Table 1 for the number of fuel assemblies in each region and Figure 1 for the Cycle 3 loading pattern.

Nominal core design parameters for Cycle 3 are as follows:

Core Power (MWt)	3411.
System Pressure (psia)	2250.
Core Inlet Temperature (°F)	546.7*
Thermal Design Flow (gpm)	365600.
Average Linear Power Density (kw/ft)	5.43

*FSAR Safety Analysis Basis Inlet Temperature is 548.2°F.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The mechanical design of the Region 5 fuel assemblies is the same as the Region 4 assemblies.* Table 1 compares pertinent design parameters of the various fuel regions. The Region 5 fuel has been designed according to the fuel performance model in Reference 3. The fuel is designed and operated so that clad flattening will not occur, as predicted by the Westinghouse model, Reference 4. For all fuel regions, the fuel rod internal pressure design basis, which is discussed and shown acceptable in Reference 5, is satisfied. It is also planned to use Wet Annular Burnable Absorber (WABA) rods, which are described and evaluated in the NRC approved WABA Evaluation Report, Reference 11.

Westinghouse has had considerable experience with Zircaloy clad fuel. This experience is extensively described in WCAP-8183, "Operational Experience with Westinghouse Cores", Reference 6. This report is updated annually.

2.2 NUCLEAR DESIGN

Cycle 3 core loading is designed to meet an $F_Q^T(z) \times P$ ECCS analysis limit of $\leq 2.237 \times K(z)$ for the current flux difference (ΔI) band width. This corresponds to the allowable ΔI-power space during normal operation using the RAOC methodology described in Reference 10.

All of the Cycle 3 values fall within the current limits except the highest boron concentration at full power.

* The fuel rod bottom end plug is changed from an external type gripper configuration to a shorter internal type gripper configuration. The resultant additional axial clearance is distributed by increasing the plenum length to increase internal pressure margin and reduce the fuel rod overall length to increase the rod growth margin. The overall length of the fuel rod is reduced by .075 inches.

The bottom nozzle uses a reconstitutable feature to facilitate bottom nozzle removal. Welded rod wires are replaced by mechanical locking cups to prevent loosening of the screws which fasten the bottom nozzle to guide thimble lower end plugs.

Table 2 provides a comparison of the Cycle 3 kinetics parameters with the current limit based on previously submitted accident analysis. These parameters are evaluated in Section 3.

Table 3 provides the control rod worths and requirements at BOL and EOL under the most limiting core conditions. The required shutdown margin is based on previously submitted accident analysis. The available shutdown margin exceeds the minimum required.

The control rod insertion limits as given in the technical specifications remain unchanged from Cycle 2.

Twenty-eight Region 5 fuel assemblies will contain fresh WABA rods arranged as shown in Figure 2. Eight Region 4 assemblies will contain clusters of twelve depleted BA's of the standard design from Cycle 2. Two symmetrically located Region 4 fuel assemblies will contain secondary source rods that were irradiated in Cycle 2. The WABA, secondary source, and depleted BA locations are shown in Figure 2.

2.3 THERMAL AND HYDRAULIC DESIGN

No significant variations in thermal margin result from the Cycle 3 reload. The present core limits, which are documented in Reference 7, were found to be conservative for Cycle 3. The core bypass flow design limit has been evaluated and is met 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, Reference 1, using the previously accepted design basis. It is concluded that the core reload will not adversely affect the ability to safely operate at 100 percent of rated power during Cycle 3. For the evaluation performed to address overpower concerns, the fuel centerline temperature limit of 4700°F can be accommodated with margin in the Cycle 3 core using the methodology described in Reference 2. The time dependent densification model, Reference 8, was used for these fuel temperature evaluations. The LOCA limit at rated power can be met by maintaining F_Q at or below 2.237.

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 all cases, it was found that the effects can be accommodated within the conservatism of the initial assumptions used in the previous applicable safety analysis. 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. Reference parameters in each of these three areas were examined as discussed below to ascertain whether new accident analyses were required.

3.2.1 Kinetics Parameters

A comparison of Cycle 3 kinetics parameters with the current limits is presented in Table 2. All parameters in Table 2 were found to be within the limiting range of values used in previous safety analysis. An evaluation of moderator feedback effects for the credible steamline break transient shows that the reactor remains subcritical.

3.2.2 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 Cycle 2 analyses.

3.2.3 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 Cycle 2 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 1 values.

Cycle 3 peaking factor and power distribution evaluations have been performed for the dropped RCCA accident according to the new dropped rod methodology described in Reference 9.

3.3 INCIDENTS EVALUATED

The Cycle 3 characteristics result in a higher critical boron concentration than assumed in the FSAR analysis for Boron Dilution at Power. An evaluation of this accident for a higher critical boron concentration at hot full power shows the analysis of the FSAR is bounding.

4.0 REFERENCES

1. Sequoyah Unit 1 Final Safety Analysis Report, USNRC Docket No. 50-327.

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

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

4. George, R. A., et.al., "Revised Clad Flattening Model," WCAP-8381, July, 1974.

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

6. Skaritka, J. and Iorii, J. A., "Operational Experience with Westinghouse Cores," WCAP-8183, Revision 12, August 1983.

7. Jones, R. G., "Reload Safety Evaluation Sequoyah Unit 1, Cycle 2," September, 1982.

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

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

10. Miller, R. W., et. al., "Relaxation of Constant Axial Offset Control (RAOC), F_Q Surveillance Technical Specification," WCAP-10217-A, August, 1982.

11. Letter from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse); Subject: Acceptance for Referencing of Licensing Topical Report WCAP-10021(P), Revision 1, and WCAP-10377 (NP), "Westinghouse Wet Annular Burnable Absorber Evaluation Report," August, 1983.

TABLE 1

SEQUOYAH UNIT 1, CYCLE 3
FUEL ASSEMBLY DESIGN PARAMETERS

Region	2	3	4	5
Enrichment (w/o U235)*	2.612	3.093	3.653	3.75
Geometric Density (percent Theoretical)*	94.5	94.4	94.6	94.5
Number of Assemblies	5	48	68	72
Approximate Burnup at Beginning of Cycle 3 (MWD/MTU) (Based on nominal Cycle 2 burnup of 11166 MWD/MTU)	23800	22700	11100	0
Approximate Burnup at nominal End of Cycle 3 (MWD/MTU) (Based on nominal Cycle 3 burnup of 14245 MWD/MTU)	35400	35700	26900	12700

* All fuel region values are as-built except Region 5 values which are nominal. An average density of 94.5% theoretical was used for Region 5 evaluations.

TABLE 2

SEQUOYAH UNIT 1, CYCLE 3
KINETICS PARAMETERS

	Previous Analysis Value Reference 1	Cycle 3 Value
Moderator Density Coefficient ($\Delta\rho$/gm/cc)	0 to 0.43	0 to <0.43
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	0.0044 to 0.0075	0.0044 to 0.0075
Maximum Prompt Neutron Lifetime (μ sec)	$\leq$ 26	$\leq$ 26
Maximum Reactivity Withdrawal Rate from Subcritical (pcm/sec)*	$\leq$ 100	$\leq$ 100
Doppler Temperature Coefficient (pcm/°F)*	-1.0 to -2.2	-1.0 to -2.2

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

TABLE 3

SEQUOYAH UNIT 1, CYCLES 2 AND 3
SHUTDOWN REQUIREMENTS AND MARGINS

	Four Loop Operation			
	Cycle 2		Cycle 3	
	BOC	EOC	BOC	EOC
Control Rod Worth (% $\Delta\rho$)				
All Rods Inserted Less Worst Stuck Rod	5.35	6.15	5.50	6.25
Less 10%(1)	4.82	5.54	4.95	5.63
Control Rod Requirements (% $\Delta\rho$)				
Reactivity Defects (Doppler, Tavg, Void, Redistribution)	1.78	3.02	1.95	3.38
Rod Insertion Allowance (RIA)	0.50	0.50	0.50	0.50
Total Requirements(2)	2.28	3.52	2.45	3.88
Shutdown Margin [(1)-(2)] (% $\Delta\rho$)	2.54	2.02	2.50	1.75
Required Shutdown Margin (% $\Delta\rho$)	1.60	1.60	1.60	1.60

Figure 1

Sequoyah Unit 1, Cycle 3

CORE LOADING PATTERN

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

X - Region Number

* Removable Fuel Rod Assembly

Region	# Assembly	w/o U-235
2	5	2.612
3	48	3.093
4	68	3.653
5	72	3.75

Figure 2
Sequoyah Unit 1 Cycle 3
Burnable Absorber and Secondary Source Locations

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2					16		16		16		16				
3			24			24		SS		24			24		
4															
5		16												16	
6			24				12*		12*				24		
7		16				12*				12*				16	
8															
9		16				12*				12*				16	
10			24				12*		12*				24		
11		16												16	
12															
13			24			24		SS		24			24		
14					16		16		16		16				
15															

X	- Number of fresh WABA
SS	Secondary Source Rods
12*	Depleted BA's from Cycle 2