RELOAD SAFETY EVALUATION

SEQUOYAH UNIT 2

CYCLE 4

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1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This report presents an evaluation for Sequoyah Unit 2, Cycle 4, which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was performed utilizing the methodology described in Reference 1 "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273-A.

All of the accidents comprising the licensing bases (Reference 2) which could potentially be affected by the fuel reload have been reviewed for the Cycle 4 design described herein.

1.2 GENERAL DESCRIPTION

The Sequoyah Unit 2, Cycle 4 [illegible] assemblies arranged in the core loading pattern configuration shown in Figure 1. During the Cycle 3/4 refueling, 9 Region 2, 48 Region 3, and 23 Region 4 fuel assemblies will be replaced with a split feed of 44 Region 6A and 36 Region 6B fuel assemblies. The 8 British Nuclear Fuels Limited (BNFL) fuel assemblies in Region 5B will remain in the core. A summary of the Cycle 4 fuel inventory is given in Table 1.

Nominal core design parameters utilized for Cycle 4 are as follows:

Core Power (MWt)	3411
System Pressure (psia)	2250
Core Inlet Temperature (°F)	546.7*
Thermal Design Flow (gpm)	365,600
Average Linear Power Density (kw/ft) (based on 144" active fuel length)	5.43

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

1.3 CONCLUSIONS

From the evaluation presented in this report, it is concluded that the Cycle 4 design does not cause any of the safety limits to be exceeded. This conclusion is based on the following:

1. Cycle 3 burnup is between 13,500 and 14,500 MWD/MTU.

2. Cycle 4 burnup is limited to a maximum of 16,220 MWD/MTU including a coastdown. Operation of a Westinghouse PWR can be continued beyond the nominal full power end of life burnup by reducing core power level. Reducing the core power provides the additional reactivity necessary to offset the effects of increased fuel burnup. The additional reactivity is gained by a reduction in moderator and Doppler feedbacks and a decrease in the xenon (fission product) nuclide concentration.

   The assumptions made by Westinghouse concerning Sequoyah Unit 2 Cycle 4 extended operation are as follows:

   a) the coastdown is achieved by power reduction at the time of initiation.
   b) the Low-low Tavg setpoint of 540°F is not changed.

3. There is adherence to plant operating limitations in the Technical Specifications.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The mechanical design of Region 6 fuel assemblies is the same as the Region 5 assemblies, except for the use of chamfered pellets, 4g rod plenum springs, and 304L stainless steel top and mid grid sleeve material.

The Region 6 pellets have a small chamfer at the outer edge of the fuel pellet ends and a reduction in the dish diameter and depth compared to the previous non-chamfered fuel pellets. The chamfer will improve pellet chip resistance during manufacturing and handling. All fuel rod design criteria are satisfied.

Also, compared to previous fuel, Region 6 fuel has a smaller rod plenum spring which satisfies a change in the non-operational 6g loading design criterion to "4g axial and 6g lateral loading with dimensional stability." Notification of Westinghouse's plans to generically incorporate this criterion change and the justification of no unreviewed safety question was previously transmitted to the NRC via Reference 11. The reduced spring force further reduces the already low potential for chamfered pellet chipping in the fuel rod.

The change in top grid sleeve material from 304 stainless steel to 304L stainless steel reduces the potential for stress corrosion cracking of the grid sleeves.

Table 1 presents a comparison of pertinent design parameters of the various fuel regions. The Region 6 fuel has been designed utilizing the Westinghouse fuel performance model (Reference 3) and the Westinghouse clad flattening model (Reference 4). The 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 (Reference 5) is satisfied for all fuel regions.

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

2.2 NUCLEAR DESIGN

Adherence to the F_Q limit is obtained by using the F_Q Surveillance Technical Specification described in Reference 7. The Cycle 4 core loading is designed to meet a $F_Q(z) \times P$ ECCS limit of $\leq 2.15 \times K(z)$.

Relaxed Axial Offset Control (RAOC) will be employed in Cycle 4 to enhance operational flexibility during non-steady state operation. The RAOC methodology and application is fully described in Reference 7. No change to the safety parameters is required for RAOC operation.

Table 2 provides a summary of the Cycle 4 kinetics characteristics compared with the current limits based on previously submitted accident analyses.

Control rod worths and requirements are provided in Table 3. The control rod worth uncertainty has been reduced from 10% in Cycle 3 to 7% in Cycle 4. Use of a 7% uncertainty on the worth of all rods less the most reactive rod stuck was discussed and shown to be acceptable in Reference 15. If the all rods out (ARO) position is redefined to be at the 222 step position or above for Cycle 4, the control rods could be slightly inserted in the active fuel. Consequently, a reactivity penalty of 0.12% $\Delta\rho$ must be taken in addition to the reactivity requirements listed in Table 3. If the ARO position is defined to be above the active fuel, then no penalty is needed. Once the ARO position is defined, it cannot be changed during the cycle of operation. In either case, the required shutdown margin, based on previously submitted accident analyses (Reference 2), exceeds the minimum required.

The core loading pattern contains 1024 wet annular burnable absorber (WABA), rods (Reference 8) located in 64 BA rod assemblies. The core loading pattern and the location of the WABA rods are shown in Figure 1.

2.3 THERMAL AND HYDRAULIC DESIGN

No significant variations in thermal margin will result from the Cycle 4 reload. The DNB core limits and safety analyses used for Cycle 4 are based on the conditions given in Sections 1.0 and 3.0. Sufficient DNB margin exists for all DNB events to meet the design criteria (References 2 and 12) for the Cycle 4 reload core.

Fuel temperatures were calculated using the revised thermal safety model described in Reference 13 and include the effects of chamfered fuel pellets. The use of chamfered fuel pellets increases the hot spot average fuel temperature less than 20°F compared to unchamfered pellets. Steady state DNBR calculations are not affected by the chamfered pellets.

For the Rod Withdrawal from Subcritical event, the core axial power distribution is severely peaked to the bottom of the core. The W-3 DNB correlation is used to evaluate DNBR in the span between the lower non-mixing vane grid and the first mixing vane grid. The DNB design basis is met. The W-3, R-grid mixing vane correlation remains applicable for the rest of the fuel assembly. Thus, the conclusions presented in the FSAR are still valid.

3.0 POWER CAPABILITY AND ACCIDENT EVALUATION

3.1 POWER CAPABILITY

The plant power capability has been evaluated considering the consequences of those incidents examined in the FSAR (Reference 2) using the previously accepted design basis. It is concluded that the core reload will not adversely affect the ability to safely operate at the design power level (Section 1.0) during Cycle 4. For the overpower transient, the fuel centerline temperature limit of 4700°F can be accommodated with margin in the Cycle 4 core. The time dependent densification model (Reference 9) was used for fuel temperature evaluations. The LOCA limit at rated power can be met by maintaining $F_Q(z)$ at or below 2.15 x K(z).

3.2 ACCIDENT EVALUATION

The effects of the reload, including the mechanical design features described in Section 2.1 and the Thermal Hydraulic design features described in Section 2.3, on the design basis and postulated incidents analyzed in the FSAR were examined. In most cases, it was found that the effects were accommodated within the conservatism of the initial assumptions used in the previous applicable safety analysis (Reference 2). For the incident which was reanalyzed, it was determined that the applicable design basis limits are not exceeded and therefore the safety conclusions of the FSAR remain valid.

A safety criterion that the reactor core remains subcritical on the soluble boron provided by the ECCS following a hypothetical large break LOCA has been evaluated for the Cycle 4 design. This criteria is met assuming the BIT remains active (20,000 PPM) and that the containment sprays are not shut off until the RWST low-low level alarm is reached (50,000 gallons) as is consistent with Table 6.3.2-5 of Reference 2.

A core reload can typically affect accident analysis input parameters in the following areas: core kinetic characteristics, control rod worths, and core peaking factors. Cycle 4 parameters in each of these three areas were examined as discussed in the following subsections to ascertain whether revisions to the accident analyses assumptions were required.

3.2.1 KINETICS PARAMETERS

Table 2 is a summary of the kinetics parameters current limits along with the associated Cycle 4 calculated values. All but the Doppler Temperature Coefficient fall within the bounds of the current limits. The Cycle 4 design Doppler Temperature Coefficient was evaluated (Section 3.4).

3.2.2 CONTROL ROD WORTHS

Changes in control rod worths may affect differential rod worths, shutdown margin, ejected rod 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 4 meets the current limit. Table 3 shows that the Cycle 4 shutdown margin requirements have been satisfied.

Cycle 4 has a normalized trip reactivity insertion rate which is slightly different than the current limit. The non-conservative deviations occur around 90 to 95 percent of rod insertion. The remaining portions of the trip insertion curve are conservative with respect to the current limit. The transients which are most sensitive to this deviation are Loss of Flow and Locked Rotor. Each of these events reach the most limiting point before the rods are inserted 90% of travel. Reactivity inserted after the most limiting point of the transient is not needed to mitigate the consequences of the event. The conclusions of the FSAR remain valid.

3.2.3 CORE PEAKING FACTORS

Peaking factors for the dropped RCCA incidents were evaluated based on the NRC approved dropped rod methodology described in Reference 10. Results show that the DNB design basis is met.

The peaking factors for steamline break have been evaluated and are within the bounds of the limits of the FSAR.

The Cycle 4 control rod ejection peaking factors were within the bounds of the Cycle 3 values, except for the end-of-life hot-zero-power case which was reanalyzed (Section 3.3).

3.3 INCIDENTS REANALYZED

The hot-zero-power end-of-life rod ejection accident was reanalyzed due to the Cycle 4 maximum F_Q exceeding the Cycle 3 value. This analysis was performed using the same methods described in references 2 and 14 however, Mode 2 and Mode 3 specific parameters were used. The Mode 2 analysis was most limiting. The results of both cases showed that the fuel rod conditions at the hot spot safisfy all the acceptance criteria specified in Reference 14. Therefore, the safety conclusions given in reference 2 remain valid.

3.4 INCIDENTS EVALUATED

The Cycle 4 design resulted in the following two changes to the assumptions in the Boron Dilution at Power parameters.

1) A higher critical boron concentration at the BOC, HFP, no Xenon, rods to insertion limits condition.

2) A greater difference between the critical boron concentration of the above condition and the N-1 rods inserted, BOC, HZP, no Xenon critical boron concentration.

Operator action time is directly proportional to a ratio of the boron concentrations used in the analysis. An evaluation using the Cycle 4 parameters demonstrated that their net effect is an increase in operator action time and thus, the current analysis is bounding.

The incidents which could be affected by an increase in the maximum negative Doppler Temperature Coefficient are Uncontrolled RCCA Bank Withdrawal at Power, Excessive Load Increase, Loss of External Electrical Load/Turbine Trip, Steamline Break Mass and Energy Release Inside Containment and Steamline Break Mass and Energy Release Outside Containment. This coefficient is used in conjunction with the Doppler Power

Coefficient to provide a correction to the Doppler defect due to fuel temperature changes resulting from core water temperature changes. Sensitivities, applicable to the current Sequoyah licensing basis analyses affected by this change, demonstrate that the Cycle 4 design Doppler temperature coefficient will not affect the transients in a manner that causes the safety conclusions of the FSAR to be invalidated.

4.0 TECHNICAL SPECIFICATION CHANGES

No Technical Specification changes are required for Cycle 4 operation. Note that Table 3.3.5 in the Technical Specification includes revised actuation times as specified in Sequoyah Nuclear Plant Technical Specification Change 75 (TVA-SQN-TS-75), which was submitted on December 23, 1986 and subsequently approved by the NRC Staff. TVA has evaluated and found acceptable the 32 second delay times listed in items 2.a and 3.a of that table for 10CFR50.46 ECCS analysis inpact.

The conclusion that the licensing basis analysis remain unaffected by the Cycle 4 reload is predicated upon the boron injection tank remaining operable according to Technical Specification 3.5.4.1.

5.0 REFERENCES

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11 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 versus 4g).

12. Letter from E. P. Rahe, Jr. (Westinghouse) to H. Berkow (NRC), NS-NRC-86-3116, dated March 25, 1986, Westinghouse Response to Additional Request on WCAP-9226-P/WCAP-9227-NP, "Reactor Core Response to Excessive Secondary Steam Releases," (Non-Proprietary).

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TABLE 1

SEQUOYAH UNIT 2 - CYCLE 4
FUEL ASSEMBLY DESIGN PARAMETERS

Region	4	5A	5B	6A	6B
Enrichment (w/o U-235)+	3.503	3.802	3.604	3.405	3.602
Density(% Theoretical)+	94.758	95.011	94.828	95.212	95.351
Number of Assemblies	45	40	28	44	36
Approximate Burnup at++ Beginning of Cycle 4 (MWD/MTU)	23800	15800	16800	0	0
Approximate Burnup at+++ End of Cycle 4 (MWD/MTU)	37800	30600	34400	19100	15900

\+ As built data.

++ Based on EOC3 = 14500 MWD/MTU

+++ Based on EOC4=16220 MWD/MTU

TABLE 2

SEQUOYAH UNIT 2 - CYCLE 4
KINETICS CHARACTERISTICS

	Current Limits	Cycle 4 Design
Most Positive Temperature Coefficient (pcm/°F)*	0	0
Doppler Temperature Coefficient (pcm/°F)*	-2.2 to -1.0	-2.9 to -1.0
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
Minimum Delayed Neutron Fraction β_{eff}, (%)	0.44	>0.44
Minimum Delayed Neutron Fraction $\beta_{eff}m$ (%) [Ejected Rod at BOL]	0.55	>0.55
Maximum Differential Rod Worth of Two Banks Moving Together (pcm/in)*	76	< 76

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

TABLE 3

SEQUOYAH UNIT 2 - CYCLE 4
END-OF-CYCLE SHUTDOWN REQUIREMENTS AND MARGINS

Control Rod Worth(%Δρ)	Cycle 3	Cycle 4
All Rods Inserted Less Worst Stuck Rod	5.86	6.07
(1) Less 10%-Cycle 3	5.28	--
Less 7%-Cycle 4	--	5.64
Control Rod Requirements		
Reactivity Defects (Doppler, T_{avg}, Void, Redistribution)	3.27	3.13
Rod Insertion Allowance	0.40	0.56
and Reposition Allowance	--	0.12
(2) Total Requirements	3.67	3.81
Shutdown Margin[(1) - (2)] (%Δρ)	1.61	1.84
Required Shutdown Margin (%Δρ)	1.60	1.60

180

		R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1						4	6B	5A	4	5A	6B	4					
2				5A	6B	6B 12	5A	6A 12	5B	6A 12	5A	6B 12	6B	5A			
3			5A	6B 16	6B 20	5A	6A 20	4	4 SS	4	6A 20	5A	6B 20	6B 16	5A		
4			6B	6B 20	5B	4	5A	6A 16	5B	6A 16	5A	4	5B	6B 20	6B		
5		4	6B 12	5A	4	6A 24	5B	4	6A 12	4	5B	6A 24	4	5A	6B 12	4	
6		6B	5A	6A 20	5A	5B	5B	6A 16	5B	6A 16	5B	5B	5A	6A 20	5A	6B	
7		5A	6A 12	4	6A 16	4	6A 16	4	6A 12	4	6A 16	4	6A 16	4	6A 12	5A	
8	90	4	5B	4	5B	6A 12	5B	6A 12	4	6A 12	5B	6A 12	5B	4	5B	4	270
9		5A	6A 12	4	6A 16	4	6A 16	4	6A 12	4	6A 16	4	6A 16	4	6A 12	5A	
10		6B	5A	6A 20	5A	5B	5B	6A 16	5B	6A 16	5B	5B	5A	6A 20	5A	6B	
11		4	6B 12	5A	4	6A 24	5B	4	6A 12	4	5B	6A 24	4	5A	6B 12	4	
12			6B	6B 20	5B	4	5A	6A 16	5B	6A 16	5A	4	5B	6B 20	6B		
13			5A	6B 16	6B 20	5A	6A 20	4	4 SS	4	6A 20	5A	6B 20	6B 16	5A		
14				5A	6B	6B 12	5A	6A 12	5B	6A 12	5A	6B 12	6B	5A			
15						4	6B	5A	4	5A	6B	4					

0

X	REGION NUMBER
Y	BA'S

SS SECONDARY SOURCES

FIGURE 1
SEQUOYAH UNIT 2, CYCLE 4
CORE LOADING PATTERN