TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

400 Chestnut Street Tower II

September 17, 1982

TVA-SQN-TS-37

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Denton:

In the Matter of) Docket No. 50-327
Tennessee Valley Authority)

In accordance with 10 CFR Part 50.59, we are enclosing 40 copies of a requested amendment to operating license DPR-77 to change the technical specifications for the Sequoyah Nuclear Plant unit 1 (Appendix A to the enclosure). The proposed amendment requests changes in the technical specifications to accommodate the unit 1 cycle 2 reload operations. The enclosure, Reload Safety Evaluation, provides a description of the changes and a justification for the changes.

In accordance with the provisions of 10 CFR Part 170.22, we have determined the proposed amendment to be Class III. This classification is based on the fact that the proposed amendment involves a single safety issue which does not involve a significant hazard consideration. The remittance of $4,000 is being wired to the Nuclear Regulatory Commission, Attention: Licensing Fee Management Branch.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

D S Kammer

D. S. Kammer
Nuclear Engineer

Sworn to and subscribed before me
this 17th day of Sept. 1982

Bryant M. Lowery
Notary Public
My Commission Expires 4/8/86

A001
s
1/40

Enclosure (40)
cc: U.S. Nuclear Regulatory Commission
Region II
Attn: Mr. James P. O'Reilly, Regional Administrator
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

8209210250p

RELOAD SAFETY EVALUATION

SEQUOYAH UNIT 1, CYCLE 2

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION AND SUMMARY

Sequoyah unit 1 is in its first cycle of operation. The unit is expected to refuel and be ready for cycle 2 startup in December 1982.

This report presents an evaluation for cycle 2 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 2 design described herein. The applicability of the current nuclear design limits was verified for cycle 2 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 2 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 1 operation is terminated between 14,600 and 15,600 MWd/t, and (2) there is adherence to plant operating limitations given in the technical specifications and their proposed modifications presented herein.

During the cycle 1/2 refueling, sixty-eight region 1 fuel assemblies will be replaced by sixty-eight region 4 assemblies. See table 1 for the number of fuel assemblies in each region and figure 1 for the cycle 2 core loading pattern.

Nominal design parameters for cycle 2 are 3411 MWt core power, 2250 psia core pressure, nominal core inlet temperature of 548.2^{o}F, and core average linear power of 5.43 kW/ft.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The mechanical design of the region 4 fuel assemblies is the same as the region 3 assemblies with the exception of minor grid modifications to minimize potential grid to grid interaction during fuel handling and a reconstitutable bottom nozzle design. In addition, the region 4 rod internal pressure has been reduced to 350 psig. Table 1 compares pertinent design parameters of the various fuel regions. The region 4 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.[(4)] For all fuel regions, the fuel rod internal pressure design basis, which is discussed and shown acceptable in reference 5, is satisfied.

Westinghouse Electric Corporation has had considerable experience with Zircaloy-clad fuel. This experience is extensively described in WCAP-8183, "Operational Experience with Westinghouse Cores."[(6)] This report is updated annually.

2.2 NUCLEAR DESIGN

Cycle 2 core loading is designed to meet an $F_Q(z) \times P$ ECCS analysis limit of $\leq 2.237 \times K(z)$. Table 2 provides a comparison of the cycle 2 kinetic characteristics with the current limit based on previously submitted accident analysis. With the exception of the least negative Doppler temperature coefficient, all of the cycle 2 values fall within the current limits. 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 analyses. The available shutdown margin exceeds the minimum required. The control rod insertion limits remain unchanged from cycle 1 as given in the technical specifications.

The PALADON Code[(7)] was used in the nuclear analyses. NRC has found this code acceptable for use on reload designs.

Twenty-eight region 4 fuel assemblies will contain fresh burnable poisons arranged as shown in figure 1. Two symmetrically located region 3 fuel assemblies will contain secondary source rods that were irradiated in cycle 1. There will also be two additional secondary source assemblies added in cycle 2 for irradiation (See figure 1 for location in region 2).

In the cycle 2 analysis, the $F^N_{\Delta H}$ limit scope was changed from 0.2 to 0.3. The change in $F^N_{\Delta H}$ with power is described by the following relationship:

$$F^N_{\Delta H} \leq 1.55 \left[1 + 0.3 \ (1-P)\right]$$

This allows an increase in allowable $F^N_{\Delta H}$ at reduced power in comparison to the previous technical specification limit while maintaining the same $F^N_{\Delta H}$ limit at full power. The increase in allowable $F^N_{\Delta H}$ at reduced power allows for optimization of the core loading pattern for full-power operation by minimizing the restriction on $F^N_{\Delta H}$ at low power. This eliminates the need to change the rod insertion limits to satisfy peaking factor criteria at low power with the control rod banks at the insertion limit. The variation in the maximum calculated $F^N_{\Delta H}$ with power with the control rods at the insertion limit for cycle 2 is shown in figure 2.

Relaxed axial offset control (RAOC) will be employed in cycle 2 to enhance operational flexibility. RAOC makes use of available margin by expanding the allowable ΔI band, particularly at reduced power. The RAOC methodology and application is fully described in reference 11. The analysis for cycle 2 indicates that no change to the safety parameters is required for RAOC operation.

Adherence to the F_Q limit is obtained by using the F_Q surveillance technical specification also described in reference 11. F_Q surveillance replaces the previous F_{xy} surveillance by comparing a measured F_Q, increased to account for expected plant maneuvers, to the F_Q limit. This provides a more convenient form of ensuring plant operation below the F_Q limit while retaining the intent of using a measured parameter to verify operation below technical specification limits. F_Q surveillance is only a change to the plant's surveillance requirements and as such has no impact on the results of the cycle 2 analysis or safety parameters.

2.3 THERMAL AND HYDRAULIC DESIGN

No significant variations in thermal margins result from the cycle 2 reload. However, the reactor core safety limits, figure 2.1.1 in the technical specifications, and the axial offset limits have been revised to reflect the increase in K from 0.2 to 0.3 in the following relationship.

$$F^N_{\Delta H} \leq 1.55 \left[1 + K \ (1-P)\right]$$

Where P = fraction of rated power for power levels less than 100 percent.

The core limits at 1775 and 2000 psia remain unchanged from the current limits. At 2250 and 2400 psia the proposed core limits are slightly more limiting below 100 percent power. The core limits have these minimal changes because, at most conditions below full power, the restriction that the average enthalpy at the vessel exit be less than the enthalpy of saturated liquid is more limiting than DNB considerations. This vessel exit enthalpy limit is not dependent on core peaking factor.

The change in axial offset limits are discussed in section 3.3.

The thermal-hydraulic methods used to analyze axial power distributions generated by the RAOC methodology are similar to those used in the constant axial offset control (CAOC) methodology. Normal operation power distributions are evaluated relative to the assumed limiting normal operation power distribution, which for Sequoyah unit 1, cycle 2, is the 1.55 cosine used in the accident analysis.

Limits on allowable operating axial flux imbalance as a function of power level from these considerations were found to be less restrictive than those resulting from LOCA F_Q considerations.

The condition II analyses were evaluated relative to the axial power distribution assumptions used to generate DNB core limits and resultant overtemperature Delta-T setpoints (including the f (ΔI) function). No changes in these limits are required for RAOC operation.

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[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 2. For the evaluation performed to address overpower concerns, the fuel centerline temperature limit of $4700^{0}F$ can be accommodated with margin in the cycle 2 core using the methodology described in reference 2. The time dependent densification model[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 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 2 parameters in each of these three areas were examined as discussed below to ascertain whether new accident analyses were required.

Kinetic Parameters

A comparison of cycle 2 kinetic 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 analyses except for the Doppler temperature coefficient (DTC). However, this change is small and, since the DTC represents only a small portion of the total negative reactivity feedback, the effect is negligible and no accidents were reanalyzed as a result. 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 2 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 2 is less than the current limit. Cycle 2 ejected rod worths were less than those used for the cycle 1 analyses, however, the hot-zero-power-end-of-life rod ejection case required reanalysis due to the peaking factors (see below).

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 referenced cycle 1 and peaking factors determined for the reload core design. In each case, it was found that the peaking factor for cycle 2 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 2 control rod ejection peaking factors were within the bounds of the cycle 1 values except for the end-of-life hot-zero-power cases which were reanalyzed (section 3.3).

Cycle 2 peaking factor and power distribuion evaluations have been performed according to the long-term methodology described in reference (9) for the dropped RCCA accident analysis.

3.3 INCIDENTS REANALYZED

The hot-zero-power end-of-life rod ejection case was reanalyzed due to the cycle 2 maximum F_Q exceeding the cycle 1 values. Table 4 gives the pertinent rod ejection parameters used in the reanalysis.

The analyses were performed using the same methods as described in references 1 and 10. The results for rod ejection show that the fuel rod conditions at the hot spot satisfies all the acceptance criteria specified in reference 10. Therefore, the safety conclusions given in reference 1 remain valid.

The change in the allowable $F_{\Delta H}^{N}$ as a function of power resulted in a change to the K constants in the overtemperature Delta-T and overpower Delta-T setpoint equations and a change to the overtemperature Delta-T $F(\Delta I)$ function.

Since the overtemperature Delta-T trip is used in the bank withdrawal at power accident, this accident was reanalyzed with the new overtemperature Delta-T setpoints. The results show that the minimum DNBR remains above the limit value. This verifies that the conclusions in reference 1 remain valid.

In the LOCA analysis, 2-percent uniform steam generator tube plugging was assumed. A total of six purge lines, four 24-inch diameter and two 12-inch diameter, were assumed to be open at the time of the accident. Initial containment temperatures used in the analysis were 105°F in the upper compartment and 125°F in the lower compartment.

4.0 Technical Specification Changes

To ensure plant operation consistent with design and safety evaluation conclusion statements made in this report and to ensure that these conclusions remain valid, several technical specification changes will be needed for cycle 2. These changes are summarized below. The changed technical specifications accompany this document (see appendix A).

Description

Incorporate the increase in K from 0.2 to 0.3 in the following relationship: $F^N_{\Delta H} \leq \left[1.55\ 1.0 + K\ (1.0\text{-}P)\right]$. The technical specification changes are as follows:

a) Replace figure 2.2-1,

b) Change Table 2.2-1 as indicated on:

page 2-7, K_1 from ≤ 1.14 to ≤ 1.15
page 2-7, K_2 from 0.009 to 0.011
page 2-8, K_3 from 0.00043 to 0.00055
page 2-10, K_6 from 0.0012 to 0.0011
page 2-8, item (i) change 30% to 29%
page 2-8, item (i) change 4% to 5%
page 2-9, item (ii) change 30% to 29%
page 2-9, ΔT trip set-point change from 0.89 to 1.5
page 2-9, item (iii) change 0.8% to 0.86%
page 2-9, item (iii) change 4% to 5%

c) Change page 3/4 2-10, Equation a R_1 relationship change 0.2 to 0.3

Change page 3/4 2-13, Equation a R_1 relationship change 0.2 to 0.3

Values in figure 3.2.3 remain unchanged,

d) Revise page B 2-1 equation for $F^N_{\Delta H}$ to $1.55\left[1 + 0.3\ (1.0\text{-}P)\right]$,

e) Replace pages B 3/4 2-2, B 3/4 2-4, B 3/4 2-5, and B 3/4 2-6.

Justification

The changes provide an increase in allowable $F^N_{\Delta H}$ at reduced power in comparison with the cycle 1 technical specifications. The increase in allowable $F^N_{\Delta H}$ at reduced power allows optimization of the core loading pattern for full power operation.

Safety Analysis

Increasing the slope of the allowable $F^N_{\Delta H}$ as a function of the power design limit from 0.2 to 0.3 requires reevaluation of the DNB protection setpoints. The setpoints for Sequoyah unit 1 cycle 2 have been updated to account for this increase in slope. The maximum calculated $F^N_{\Delta H}$ through the power range of Sequoyah unit 1 cycle 2 has been verified to be less than the value allowed with the 0.3 $F^N_{\Delta H}$ slope multiplier. The effect on specific parameters is discussed in this report.

Description

Incorporate RAOC methodology for power distribution control into the Sequoyah technical specifications. The attached changes are as follows:

a) Replace sections 3.2.1, 4.2.1.1, 4.2.1.2, B 3/4.2, and B 3/4.2.1,

b) Replace figure 3.2-1,

c) Delete sections 4.2.1.3 and 4.2.1.4,

d) Delete figure B 3/4 2-1.

Reference

1. R. W. Miller, et.al.; Relaxation of Constant Axial Offset Control For Sequoyah Unit 1, Cycle 2; August 1982.

2. Millstone Nuclear Plant, unit 2, cycle 4 SER, Amendment 61, October 6, 1980.

Justification

In a plant incorporating RAOC operation, the technical specifications are modified to remove all references to CAOC in section 3/4.2.1 and the corresponding bases. RAOC application has the following advantages:

a) Maneuvering capability is enhanced and boron system duty can be minimized or smoothed,

b) Operator action required to conform to power distribution technical specifications is reduced because rod motion corrections are reduced,

c) Return to power capability after a trip is greatly increased.

Safety Analysis

The RAOC methodolgy utilizes the plant-specific LOCA and DNB margin to set the allowable ΔI band. Limits on allowable operating axial flux imbalances as a function of power level considering limiting condition I power distributions were found to be less restrictive than those resulting from LOCA F_Q considerations. Condition II analyses were evaluated relative to the axial power distribution assumptions used to generate DNB core limits. No changes in these limits are required for RAOC operation.

The RAOC methodology is similar to CAOC methodology with the following exception.

> The method used for generating the xenon shape library is different. Previously, a library based upon xenon oscillation studies was used. For cycle 2, Westinghouse generates a xenon parameter range library and systematically reconstructs the xenon distribution when needed. In both methods, the entire range of xenon and rod insertion limits are covered.

A detailed description of RAOC is included in reference 1.

Description

Delete the last sentence of action A of Limiting Condition for Operation 3.2.2. This deletes the requirement for going to hot standby to reduce the overpower Delta-T trip setpoint with F_Q exceeding its limit.

Justification

The overpower Delta-T trip setpoint can be reduced one channel at a time while at power. It is not necessary to go to hot standby to make these setpoint changes.

Safety Analysis

The purpose of this action statement is to compensate for a measured $F_Q(z)$ exceeding its limit by reducing the overpower Delta-T setpoint. Reducing this setpoint provides a more conservative reactor trip. This action coupled with required power reduction and reduction of power range neutron flux high trip setpoint ensures FSAR assumptions remain valid should an accident occur under these conditions.

Description

Replace $F_{xy}(z)$ surveillance currently in Sequoyah technical specifications with $F_Q(z)$ surveillance. The attached changes are as follows:

a) Sections 3.2.2, 4.2.2.2, 4.2.2.3, and 6.9.1.14 are replaced,

b) The appropriate bases are changed (B 3/4.2).

Reference

R. W. Miller, et.al.; Relaxation of Constant Axial Offset Control For Sequoyah Unit 1, Cycle 2; August 1982.

Justification

$F_{xy}(z)$ is implicitly included in the $F_Q(z)$ measurement, and the intent of the technical specification is to monitor $F_Q(z)$ using a measured parameter. Therefore, the $F_{xy}(z)$ surveillance requirements in the technical specifications are replaced with $F_Q(z)$ surveillance. $F_Q(z)$ surveillance provides the following advantages:

a) Credit can be taken for the actual power distribution (and resulting $F_Q(z)$ values) measured in the plant.

b) Monitoring $F_Q(z)$ and increasing the value for expected plant maneuvers provides for a more convenient form of ensuring plant operation below the $F_Q(z)$ limit.

c) The cycle dependent factors will be reported in a peaking factor report which will reduce technical specification changes. A description of the peaking factor report is included in section III.B.2 of the reference.

Safety Analysis

$F_Q(z)$ surveillance implicitly includes $F_{xy}(z)$ and retains the use of a measured parameter to verify operation below the technical specification limits. F_Q surveillance is only a change to the plants' surveillance requirements and as such has no impact on the results of the cycle 2 analyses or safety parameters. A detailed description of the $F_Q(z)$ surveillance is included in section III.B.1 of the reference.

Description

For limiting condition for operation 3.6.1.5, change the upper limits for the upper and lower containment air temperatures to 105°F and 125°F, respectively.

Justification

The Sequoyah LOCA analysis has been repeated with the upper limits of the containment upper and lower compartment air temperatures at 105°F and 125°F, respectively.

Safety Analysis

The upper limit on containment air temperature ensures that the containment air mass is limited to an initial air mass sufficiently high so that blowdown of the reactor coolant system (RCS) subsequent to a LOCA is consistent with analytical assumptions. The new LOCA analysis shows that the conclusions presented in the FSAR are still valid and the peak clad temperature remains below 2200°F.

Description

For limiting condition for operation 3.6.1.9, change the number of purge supply and exhaust lines allowed open to three pairs.

Justifications

The Sequoyah LOCA analysis has been repeated with the supply and exhaust lines to the upper and lower containment and the instrument room all open at the initiation of the LOCA.

The Tennessee Valley Authority (TVA) has also assessed the site boundary dose subsequent to a LOCA with seven (7) 24-inch purge lines opened.

Safety Analysis

The new LOCA analysis shows that the conclusions presented in the FSAR with respect to reactor coolant system (RCS) blowdown are still valid. Peak clad temperature remains below 2200°F. Further, the TVA assessment of the site boundary dose subsequent to a LOCA shows the limits of 10 CFR 100 are met.

Description

Remove unnecessary statement in the bases describing quadrant power tilt ratio (section B 3/4.2.4).

Justification

The paragraph above this statement defines the purpose for the limit, therefore, this additional statement is unnecessary.

Safety Analysis

There are no safety implications.

5.0 DROPPED ROD ACCIDENT ANALYSIS - REMOVAL OF OPERATING RESTRICTIONS

Our July 22, 1982 letter to Ms. E. Adensam (see appendix D) formally requested your staff review the material submitted to you as NS-EPR-3545, January 20, 1982, and subsequently remove the interim operational restrictions before startup of Sequoyah unit 1, cycle 2.

We believe the removal of these operating restrictions is justified and request your concurrence.

6.0 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. Jones, R. G. and Iorii, J. A., "Operational Experience with Westinghouse Cores," WCAP-8183 Revision 11, May 1982.

7. Camden, T. M., et. al., "PALADON - Westinghouse Nodal Computer Code," WCAP-9486, December 1978.

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

9. Letter; Rahe (Westinghouse) to Berlinger (NRC), "Dropped Rod Methodology for Negative Flux Rate Trip Plants" NS-EPR-2545, January 20, 1982.

10. Risher, D. H., "An Evaluation of the Rod Ejection Accident in Westinghouse PWR's Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975.

11. Letter; Rahe (Westinghouse) to Berlinger (NRC), "Relaxation of Constant Axial Offset Control (RAOC)," NS-EPR-2649, August 31, 1982.

TABLE 1

FUEL ASSEMBLY DESIGN PARAMETERS

SEQUOYAH UNIT 1 - CYCLE 2

Region	1	2	3	4
Enrichment (w/o U235)*	2.10	2.61	3.09	3.65
Geometric Density (percent Theoretical)*	94.5	94.5	94.4	94.5
Number of Assemblies	5	72	48	68
Approximate Burnup at Beginning of Cycle 2 (MWD/MTU)	14500	16600	10200	0

*All fuel regions except region four are as-built values: region four values are nominal. An average density of 94.5% theoretical was used for region 4 evaluations.

TABLE 2

KINETICS CHARACTERISTICS

SEQUOYAH UNIT 1 - CYCLE 2

	Previous Analysis Value (1) (7)	Cycle 2 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	.0044 to .0075	.0044 to .0075
Maximum Prompt Neutron Lifetime (μsec)	$\leq$26	$\leq$26
Maximum Reactivity Withdrawal Rate from Subcritical (pcm/sec)*	$\leq$100	$\leq$100
Doppler Temperature Coefficient (pcm/^{o}F)*	-1.4 to -2.9	-1.0 to -2.9

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

TABLE 3

SHUTDOWN REQUIREMENTS AND MARGINS

SEQUOYAH UNIT 1 - CYCLE 1 AND 2

	Four-Loop Operation			
	Cycle 1		Cycle 2	
	BOC	EOC	BOC	EOC
Control Rod Worth ($\%\Delta\rho$)				
All Rods Inserted Less Worst Stuck Rod	6.61	6.18	5.35	6.15
Less 10%(1)	5.95	5.56	4.82	5.54
Control Rod Requirements ($\%\Delta\rho$)				
Reactivity Defects (Doppler, Tavg, Void, Redistribution)	2.16	2.94	1.78	3.02
Rod Insertion Allowance (RIA)	0.50	0.50	0.50	0.50
Total Requirements2	2.66	3.44	2.28	3.52
Shutdown Margin (1)-(2) ($\%\Delta\rho$)	3.29	2.12	2.54	2.02
Required Shutdown Margin ($\%\Delta\rho$)	1.60	1.60	1.60	1.60

TABLE 4

ROD EJECTION PARAMETERS

FOUR-LOOP OPERATION

SEQUOYAH UNIT 1

	Previous Analysis Values(1)	Cycle 2 Values	Used in Analysis
HZP-EOL			
Max Ejected Rod Worth, $\%\Delta\rho$	0.98	0.565	0.565
Max F_Q^N	19.1	25.3	25.3
Min Beff	.0044	.0044	.0044

HZP - Hot Zero Power EOL - End of Life

CORE LOADING PATTERN

Sequoyah Unit 1, Cycle 2

180°

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

270° (left side) · 90° (right side, at row 8)

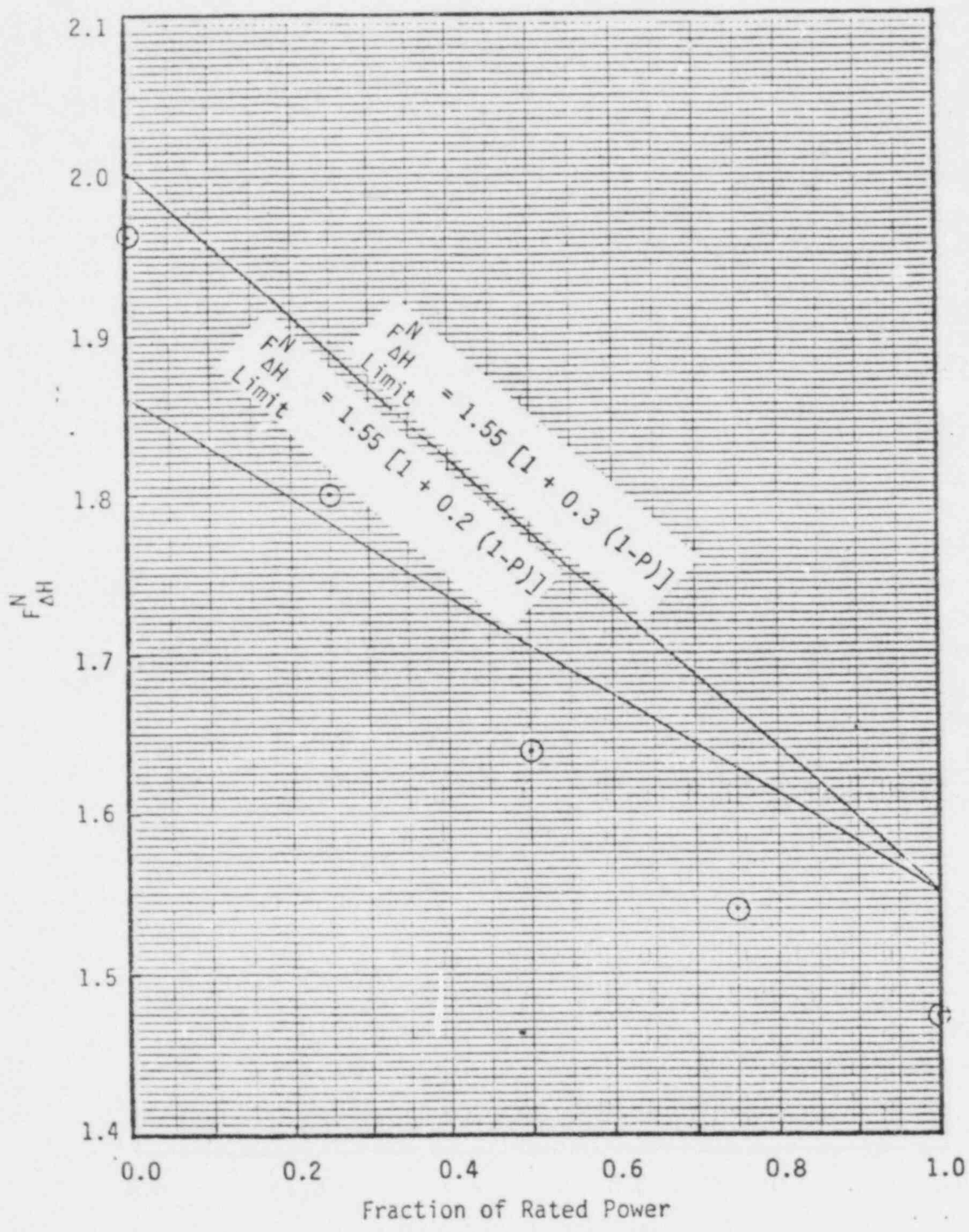
0°

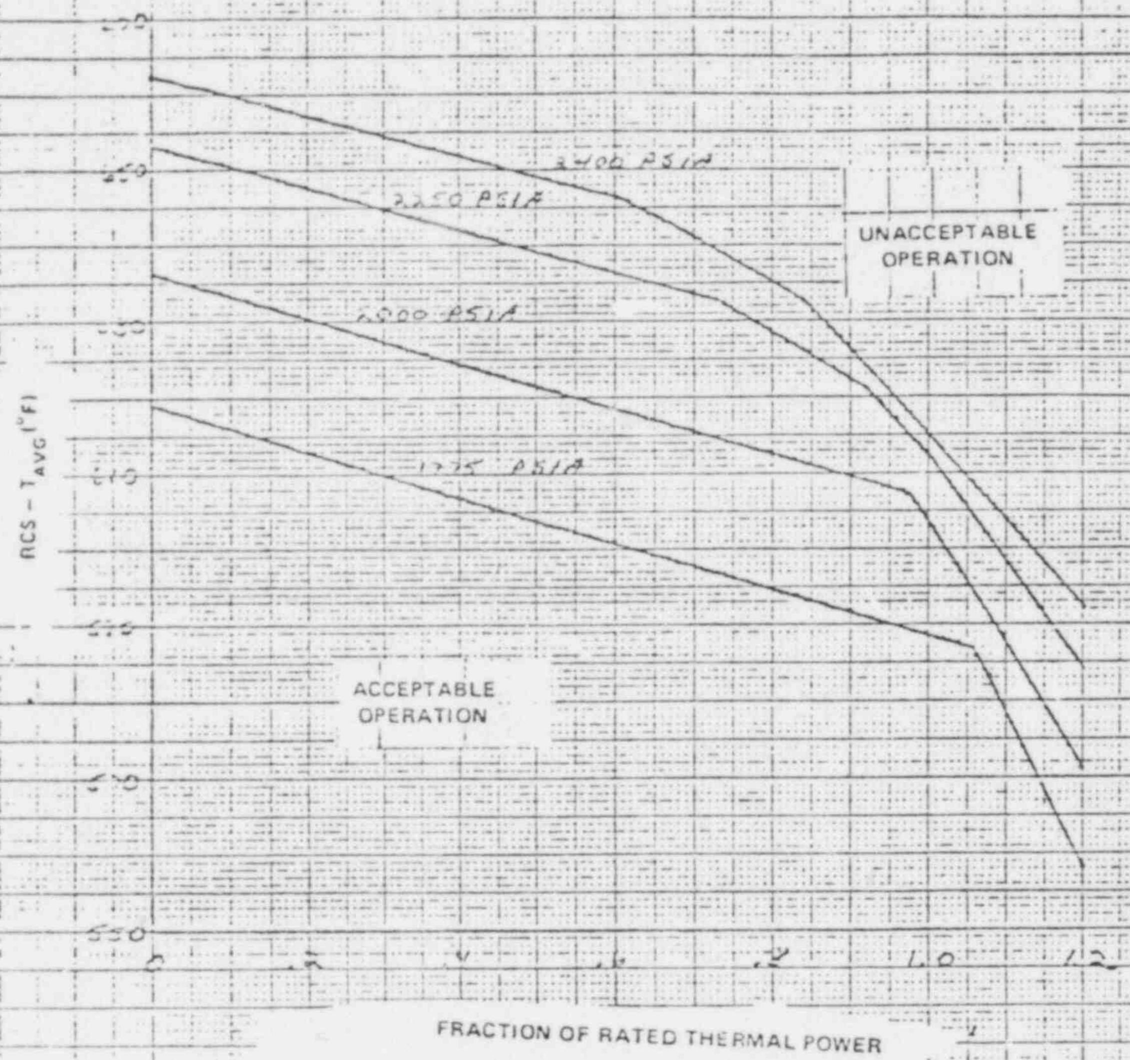
X Y SS	- Region Number - Number of Burnable Poison Rods - Secondary Source Rods

Figure 1

FIGURE 2

MAXIMUM CALCULATED VALUES OF $F^N_{\Delta H}$
WITH RESPECT TO TECHNICAL SPECIFICATION LIMITS





Revised Figure 2.1-1. Reactor Core Safety Limit – Four Loops in Operation

FIGURE 3