RELOAD SAFETY EVALUATION
INDIAN POINT NUCLEAR PLANT
UNIT 3, CYCLE 6

JULY 1987

Edited by
J. Skaritka
F. J. Silva

Approved: Edward A. Dzenis
E. A. Dzenis, Manager
Core Operations
Commercial Nuclear Fuel Division

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

The Indian Point Nuclear Power Plant, Unit Number 3 has completed its fifth cycle of operation. The unit is expected to be refueled and ready for Cycle 6 startup in July 1987.

This report presents an evaluation for Unit 3 Cycle 6, which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was accomplished utilizing the methodology described in WCAP-9273-A, "Westinghouse Reload Safety Evaluation Methodology"(1).

Indian Point 3 operated in Cycle 5 with 117 Westinghouse 15x15 low parasitic (LOPAR) fuel assemblies and 76 Westinghouse optimized fuel assemblies. For Cycle 6 and subsequent cycles, it is planned to refuel the Indian Point 3 core with Westinghouse 15x15 optimized fuel assembly (OFA) regions. The NRC reviewed the licensing submittal (2) and approved the request for the transition from LOPAR fuel to OFA and associated proposed changes to the Indian Point 3 Technical Specifications. The licensing submittal justifies the compatability of OFAs with LOPAR fuel assemblies in a mixed-fuel core as well as operating with a full OFA core. The licensing submittal contains mechanical, nuclear, thermal-hydraulic, and accident evaluations which are applicable to the Cycle 6 safety evaluation. The NRC also approved a supporting evaluation report (3) which shows that plant operating limitations can be satisfied with an average steam generator tube plugging level of 24% with a maximum of 30% in any loop providing the following conditions are met:

1. The reactor vessel flow must be equal to or greater than 323,600 gpm.

2. During steady state operation at full power, the hottest cold leg inlet temperature must not exceed 542.9°F plus 2°F for control deadband.

3. The overtemperature delta-T reactor trip channels must be calibrated during power operation in terms of both delta-T and T_{avg} indicated by each channel at nominal full power.

All of the accidents comprising the FSAR licensing bases which could potentially be affected by the fuel reload have been reviewed for the Cycle 6 design described herein. The results of new analyses are included, and the justification for the applicability of previous accident analysis results for the Cycle 6 accident evaluations is presented.

1.2 GENERAL DESCRIPTION

The Indian Point 3 Cycle 6 reactor core is comprised of 193 fuel assemblies arranged in the core loading pattern configuration shown in Figure 1. During the Cycle 5/6 refueling, 76 fuel assemblies will be replaced with Regions 8-A and 8-B fresh fuel. A summary of the Cycle 6 fuel inventory is given in Table 1.

Nominal core design parameters utilized for Cycle 6 are as follows:

Core Power (Mwt)	3025 (100% rated)
System Pressure (psia)	2250
Core Inlet Temperature (°F)	542.9*
Thermal Design Flow (gpm)	323,600*
Average Linear Power Density (kw/ft) (based on best estimate hot, densified core average stackheight of 143.9 inches)	6.24

* Accounts for an average of 24% steam generator tube plugging, maximum of 30% in any steam generator.

1.3 CONCLUSIONS

From the evaluation presented in this report, it is concluded that the Cycle 6 design does not result in the previously acceptable safety limits for any incident to be exceeded. This conclusion is based on the following:

1. Cycle 5 actual burnup of 14,263 MWD/MTU.

2. Cycle 6 burnup will not exceed 15,250 MWD/MTU, which includes a power coastdown.

3. There is adherence to plant operating limitations as given in the Technical Specifications; changes are not needed as a result of the Cycle 6 design.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The new Regions 8-A and 8-B fuel assemblies are Westinghouse 15x15 OFAs. The mechanical design of the Region 8 fuel assemblies is the same as the Region 7 assemblies, except for 4g rod plenum springs, 304L stainless steel top grid sleeve material and a radiused design fuel rod bottom end plug. Justification of OFA compatability with the Westinghouse 15x15 LOPAR fuel assemblies in a mixed core is presented in the licensing submittal.(2)

Compared to previous fuel, Region 8 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." The reduced spring force further reduces the already low potential for chamfered pellet chipping in the fuel rod.

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

The fuel rod bottom end plug is changed from a chamfered end to a radiused-end to improve rod loading and reduce the potential of grid damage during rod loading.

Table 1 presents a comparison of pertinent design parameters of the various fuel regions. The Region 8-A and 8-B fuel has been designed utilizing the Westinghouse fuel performance model(5) and the Westinghouse clad flattening model.(6) The Westinghouse 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(7) is satisfied for all fuel regions.

Westinghouse has considerable experience with Zircaloy clad fuel which includes OFAs. This experience is described in the report, "Operational Experience with Westinghouse Cores."(8)

2.2 NUCLEAR DESIGN

The Cycle 6 core loading satisfies the F_Q^T x P ECCS limit of $\leq$ 2.20 x K(z), given in Figure 2, which is necessary to meet the current PCT required by the NRC. Approved control rod insertion limits (Figure 3) have changed from rod limits used for the Cycle 5 RSE. In addition, the flux difference (ΔI) bandwidth during normal operation is unchanged from the Cycle 5 $\pm$5% ΔI.

Table 2 provides a comparison of the Cycle 6 kinetics characteristics with the current analysis value based on previously submitted accident analysis. It can be seen from the table that some Cycle 6 kinetics parameter values fall outside the range of the previous analysis values. These parameters are evaluated in Section 3.0.

Table 3 provides the control rod worths and requirements at the most limiting condition during the cycle. The available shutdown margin exceeds the minimum required. Note that the rod insertion allowances at BOC and EOC are the as-calculated value.

Thirty-six Region 8-A and thirty-two Region 8-B and two Region 7-B fuel assemblies contain fresh wet annular burnable absorber rods. No depleted burnable absorber rods are used. Two Region 7-A fuel assemblies contain secondary source rod assemblies. See Figure 1 for the location of burnable absorber and source rods.

The minimum refueling boron concentration required to maintain at least 10 percent shutdown margin for Cycle 6 is 1941 ppm. This is a lower refueling boron concentration than that assumed in the FSAR analysis of the Boron Dilution During Refueling event. Therefore, this event was reanalyzed as discussed in Section 3.3.

2.3 THERMAL AND HYDRAULIC DESIGN

No variation of thermal margins resulted from the Cycle 6 mixed OFA and LOPAR fueled core. The present core safety limits in the technical specifiations are conservative for the Cycle 6 reload core. Sufficient margin exists for all DNB events to meet the design criteria[4,9] for the Cycle 6 reload core.

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 using the previously accepted design basis. It is concluded that the core reload will not adversely affect the ability to safely operate at 100% of rated power during Cycle 6. For the overpower transient, the fuel centerline temperature limit of 4700°F can be accommodated with margin during Cycle 6. The time dependent densification model[10] was used for fuel temperature evaluations. The LOCA limit is met by maintaining $F_Q \times P$ at or below 2.20 x K(Z) given in Figure 2. This limit is satisfied for the power control maneuvers allowed by the technical specifications, which assures that the final acceptance criteria (FAC) limits are met for a spectrum of small and large LOCAs.

3.2 ACCIDENT EVALUATION

The effects of the reload on the design basis and postulated incidents analyzed in the FSAR[4] were 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.

An evaluation has demonstrated that the Cycle 6 reactor and borated water sources ensure core subcriticality following a postulated Large Break LOCA, thus satisfying the post-LOCA long term core cooling requirements.

A core reload can typically affect accident analysis input parameters in the following areas: core kinetics characteristics, control rod worths, and core peaking factors. Cycle 6 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 6 core physical parameters with the previous analysis values is presented in Table 2. The most negative moderator temperature coefficient (rodded) for Cycle 6 is -38 pcm/°F compared to the previous analysis value of -35 pcm/°F. This parameter is considered in the analysis of incidents where the maximum reactivity feedback is used to increase the severity of the event for conservatism. All accident analysis that consider the use of the most negative moderator temperature coefficient have used a value more conservative than the Cycle 6 value with the exception of the Startup of an Inactive Loop event, the Decreased Enthalpy event, and the Excessive Load Increase event. The current analyses for all three of these events are those presented in the original FSAR which used a value of -35 pcm/°F. An evaluation of the impact of the 3 pcm/°F difference in the negative moderator temperature coefficient was made and determined to have a negligible effect on the FSAR results. There is sufficient margin to the DNBR limit to accommodate the more negative moderator temperature coefficient for these FSAR events. Therefore, no reanalysis is required.

The most negative Doppler temperature coefficient is -2.50 pcm/°F compared to the previous value of -1.97 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.

Like the most negative moderator temperature coefficient discussed above, the most negative Doppler temperature coefficient is conservatively used in the analyses of incidents where maximum reactivity feedback is considered. From a review of these incidents, it was found that only the maximum reactivity feedback cases for the Loss of External Electrical Load incident reported in the FSAR assumed a negative Doppler temperature coefficient greater than the Cycle 6 value of -2.50 pcm/°F. Therefore, this incident was reanalyzed and is discussed in Section 3.3.

The least negative Doppler - Only Power Coefficient as a function of power is -9.154 pcm/% power (constant) for Cycle 6 compared to the previous constant value of -11.84 pcm/% power. In prior cycles, the previous analysis value had been -7.0 pcm/% power (constant) and conservatively bounded all incidents considering a least negative value. Subsequent to the Cycle 5 reload evaluation, but prior to Cycle 5 operation, a safety evaluation for operation with Weed RTDs[11] was performed. Part of the basis of this evaluation included an analysis of the Uncontrolled Control Rod Assembly (RCCA) Withdrawal at Power event. In this analysis of the minimum reactivity feedback cases, a value of -11.84 pcm/% power was assumed. To bound the Cycle 6 reload value, the minimum reactivity feedback cases for the uncontrolled RCCA Withdrawal at Power event was reanalyzed using a constant value of -7.0 pcm/% power. This reanalysis is discussed in Section 3.3.

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 6 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 6 does not exceed the current limit. Cycle 6 ejected rod worths are within the bounds of the current limits.

3.2.3 Core Peaking Factors

Evaluation of peaking factors for the rod out of position and dropped bank incidents show that the minimum DNBR criteria is satisfied. However, due to higher Cycle 6 peaking factors for the dropped bank incident, a reanalysis of this accident was required to show that the minimum DNBR criterion was satisfied, as discussed in Section 3.3.

The steamline break transients (FSAR Section 14.2.5) were evaluated for Cycle 6. The evaluations showed that the Cycle 6 peaking factors are within the bounds of the previous analysis, and DNBR limits (See Section 2.3) are satisfied.

3.3 INCIDENTS REANALYZED

As indicated above, several reanalyses were performed due to non-conservative Cycle 6 values when compared to the current limit values. The transients reanalyzed are discussed below.

Uncontrolled RCCA Withdrawal at Power

This event is described in Section 14.1.2 of the FSAR. The analysis is performed for a range of reactivity insertion rates from different initial power levels to verify the adequacy of the reactor protection system to prevent DNB and to prevent water relief through the pressurizer safety and relief valves. The complete range of cases was reanalyzed for minimum reactivity feedback with the assumption of a constant doppler-only power coefficient of -7.0 pcm/% power.

The results of the analysis show that the combination of the power range high neutron flux, overtemperature delta-T and high pressurizer water level reactor trips provide the necessary protection against a rod withdrawal at power event. For all cases, the DNB design basis is met and there is no water relief through the pressurizer relief or safety valves. Thus there is no adverse impact on the core or reactor coolant system integrity, and the conclusions of the FSAR remain valid.

Loss of External Electrical Load

This event is described in Section 14.1.8 of the FSAR. The analysis is performed to verify the adequacy of the reactor protection system and pressurizer safety valves to prevent DNB and overpressurization of the RCS and the main steam system. This event was reanalyzed for maximum reactivity feedback with the assumption of a Doppler temperature coefficient of -2.90 pcm/°F.

The results of the analysis show that a loss of external electrical load presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices in the two systems are adequate to limit the maximum pressures to within 110 percent of the design values. The integrity of the core is maintained by operation of the reactor protection system, i.e., the minimum DNBR is maintained above the limit value.

Boron Dilution during Refueling

As noted in Section 2.2, the minimum refueling boron concentration required to maintain at least 10 percent shutdown margin for Cycle 6 is 1941 ppm. This refueling boron concentration is less than the 2100 ppm value assumed in the analysis of this event as reported in FSAR Section 14.1.5. With a lower refueling boron concentration, the time to dilute from the refueling concentration to the critical boron concentration of 1190 ppm (assumed in the FSAR) is shorter than that calculated using the values reported in the FSAR. Thus, this event was reanalyzed using a lower initial refueling boron concentration value of 1800 ppm, and the time to dilute to 1190 ppm was determined. With all other assumptions the same as noted in the FSAR (i.e., minimum RCS volume = 4456 $ft.^3$, maximum dilution flow rate conservatively set at 300 gpm), the resulting dilution time is 45.8 minutes. Since the criterion for this event is 30 minutes, there is sufficient time available for the operator to determine the cause of dilution and isolate the primary water makeup source by closing valves.

Rod Cluster Control Assembly (RCCA) Dropped Bank

As indicated in Section 3.2.3, the Dropped RCCA Bank incident (e.g., core power reductions below turbine runback setpoint) was reanalyzed for Cycle 6. This reanalysis was necessary due to higher Cycle 6 peaking factors ($F^N_{\Delta H}$) for RCCA bank worths combined with the application of older, overly conservative, worst condition statepoint methodology no longer required. This reanalysis of the Dropped RCCA Bank incident used the latest Dropped RCCA Rod/Bank methodology for turbine runback plants consistent with the existing Dropped RCCA Rod analysis.

In this Dropped RCCA Bank analysis, RCCA bank worths from 200 pcm to 1600 pcm were used to determine the peak core heat flux reached during the transient (as a function of worth) after the core power falls below the turbine runback setpoint (74% power including uncertainty).

These peak heat flux values, along with the peaking factors associated with the corresponding bank worths and system conditions that bound the transient (e.g., power, flow, pressure, coolant temperature), were then used in the DNBR evaluation of this event. The results of this analysis show that the minimum DNBR criteria is met.

4.0 REFERENCES

1. Bordelon, F.M., et. al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273-A, July 1985.

2. Letter to NRC from J. P. Bayne (PASNY), "Technical Specification Changes Regarding Cycle 4-5 Refueling," April 1, 1985.

3. Segletes, J., et. al., "Safety Evaluation for Indian Point Unit 3 with Asymmetric Tube Plugging Among Steam Generators," WCAP-10704, Revision 1, January, 1986.

4. Final Safety Analysis Report - Indian Point Unit 3, Docket Number 50-286, Updated July 1986.

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

6. George, R. A. et. al., "Revised Clad Flattening Model," WCAP-8377 (P) and WCAP-8381 (NP), July 1974

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

8. Letter from E. P. Rahe, Jr. (Westinghouse to J. Lyons (NRC), Subject: Transmittal of "Operational Experience with Westinghouse Cores (through December 31, 1985)," NS-NRC-86-3184, November 26, 1986.

9. 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 Release," (Non-Proprietary).

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

11. Meeuwis, O., "RCS Weed RTD Error Allowances Safety Evaluation for Cycle 5 Operation," INT-85-641, August 26, 1985.

12. Tuttle, B. J. (Ed.), "Reload Safety Evaluation - Indian Point Unit 3, Cycle 5," March 1985.

TABLE 1

FUEL ASSEMBLY DESIGN PARAMETERS
INDIAN POINT UNIT NUMBER 3 - CYCLE 6

Region	5	6-A	6-B	7-A	7-B	8-A	8-B
Enrichment (w/o of U 235)*	3.301	3.201	3.397	3.198	3.402	3.205	3.598
Density (percent theoretical)*	94.66	94.37	94.41	94.92	94.97	95.04	94.89
Number of Assemblies	4	2	36	36	39	44	32
Burnup at Beginning of Cycle 6 (MWD/MTU)+	28,135	29,186	27,374	15,891	15,774	0	0
Fuel Type	LOPAR	LOPAR	LOPAR	OFA	OFA	OFA	OFA

*All fuel region enrichments and densities are as-built values.

+Based on a actual Cycle 5 burnup of 14,263 MWD/MTU.

TABLE 2

KINETICS CHARACTERISTICS

INDIAN POINT UNIT NUMBER 3 - CYCLE 6

	Previous Analysis Values (3), (4), (12)	Cycle 6
Moderator Temperature Coefficient, (PCM/°F)*	-35 to 0.0	-38 to 0.0**
Least Negative Doppler - Only Power Coefficient, Zero to Full Power (pcm/% power)*	-11.84 (constant)	-9.154 (constant)
Most Negative Doppler - Only Power Coefficient Zero to Full Power (pcm/% power)*	-14.27 (constant)	>-14.27 (constant)
Delayed Neutron Fraction β_{eff} (percent)	0.44 to 0.70	0.44 to 0.70
Maximum Prompt Neutron Lifetime (μ sec)	19	$\leq$19
Maximum Reactivity Insertion Rate for Two Banks Moving Together at HZP (pcm/sec)*	80	$\leq$80
Doppler Temperature Coefficient (pcm/°F)	-1.97 to -1.4	-2.50 to -1.4

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

**The moderator temperature coefficient is predicted to be negative at all normal operating conditions. In the physics test condition of HZP-ARO, the moderator coefficient is predicted to be positive at beginning of life. The coefficient is predicted to be negative, however, with the expected use of control rods during the physics tests.

TABLE 3

SHUTDOWN REQUIREMENTS AND MARGINS
INDIAN POINT UNIT NUMBER 3 - CYCLES 5 AND 6

	Cycle 5		Cycle 6	
	BOC	EOC	BOC	EOC
Control Rod Worth (percent $\Delta\rho$)				
All Rods Inserted Less Worst Stuck Rod	6.91	7.13	6.29	7.10
(A) Less 10%	6.22	6.42	5.66	6.39
Control Rod Requirements (percent $\Delta\rho$)				
Reactivity Defects (Doppler, Tavg, Void, Redistribution)	1.89	2.76	1.86	2.68
Rod Insertion Allowance	0.50	0.60	0.50	0.50
(B) Total Requirements	2.39	3.36	2.36	3.18
Shutdown Margin [(A)-(B)] (percent $\Delta\rho$)	3.83	3.06	3.30	3.21
Required Shutdown Margin (percent $\Delta\rho$)(1)	1.0	1.72	1.0	1.72

90

		15	14	13	12	11	10	9	8	7	6	5	4	3	2	1	
	R					7-2	7-1	8-1	7-2 12	8-1	7-1	7-2					
	P			6-2	7-1	8-2 4	7-2	8-2 12	7-2	8-2 12	7-2	8-2 4	7-1	6-2			
	N		5	7-2	8-2 12	7-1	8-2 12	6-2	8-1 12	6-2	8-2 12	7-1	8-2 12	7-2	5		
	M		7-1	8-2 12	7-2	8-1 12	6-2	8-1 12	6-2	8-1 12	6-2	8-1 12	7-2	8-2 12	7-1		
	L	7-2	8-2 4	7-1	8-1 12	6-2	8-1 12	7-1	7-2	7-1	8-1 12	6-2	8-1 12	7-1	8-2 4	7-2	
	K	7-1	7-2	8-2 12	6-2	8-1 12	6-2	8-1 12	6-2	8-1 12	6-2	8-1 12	6-2	8-2 12	7-2	7-1 SS	
	J	8-1	8-2 12	6-2	8-1 12	7-1	8-1 12	7-1	7-2	7-1	8-1 12	7-1	8-1 12	6-2	8-2 12	8-1	
0	H	6-1	7-2	8-1 12	6-2	7-2	6-2	7-2	7-2	7-2	6-2	7-2	6-2	8-1 12	7-2	6-1	180
	G	8-1	8-2 12	6-2	8-1 12	7-1	8-1 12	7-1	7-2	7-1	8-1 12	7-2	8-1 12	6-2	8-2 12	8-1	
	F	7-1 SS	7-2	8-2 12	6-2	8-1 12	6-2	8-1 12	6-2	8-1 12	6-2	8-1 12	6-2	8-2 12	7-2	7-1	
	E	7-2	8-2 4	7-1	8-1 12	6-2	8-1 12	7-1	7-2	7-1	8-1 12	6-2	8-1 12	7-1	8-2 4	7-2	
	D		7-1	8-2 12	7-2	8-1 12	6-2	8-1 12	6-2	8-1 12	6-2	8-1 12	7-2	8-2 12	7-1		
	C		5	7-2	8-2 12	7-1	8-2 12	6-2	8-1 12	6-2	8-2 12	7-1	8-2 12	7-2	5		
	B			6-2	7-1	8-2 4	7-2	8-2 12	7-2	8-2 12	7-2	8-2 4	7-1	6-2			
	A					7-2	7-1	8-1	7-2 12	8-1	7-1	7-2					

270

KEY:

RN B/SS	RN = FUEL REGION NUMBER B = NUMBER OF BURNABLE ABSORBERS SS = SECONDARY SOURCE RODS

FIGURE 1

INDIAN POINT UNIT 3, CYCLE 6 LOADING PATTERN

FORM PPLAIN

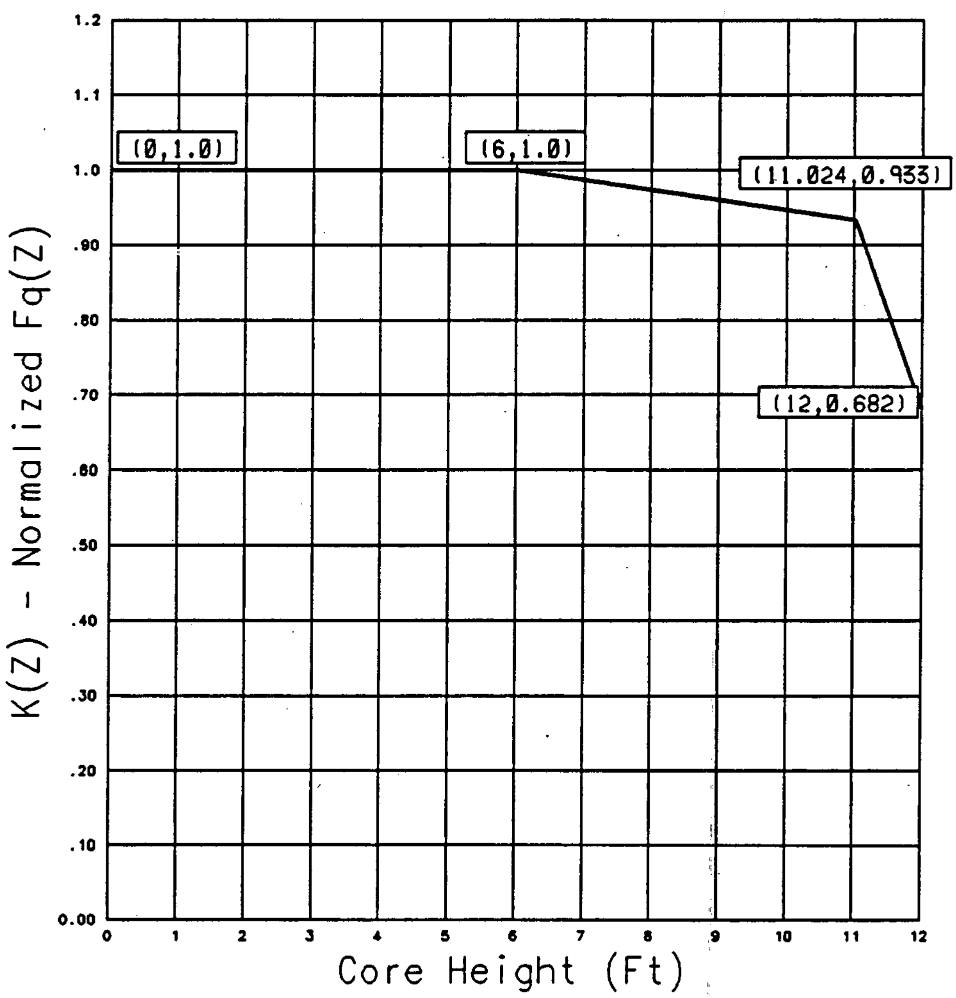

FIGURE 2

MAXIMUM $F_Q^T * P_{REL}$ VS. AXIAL CORE HEIGHT

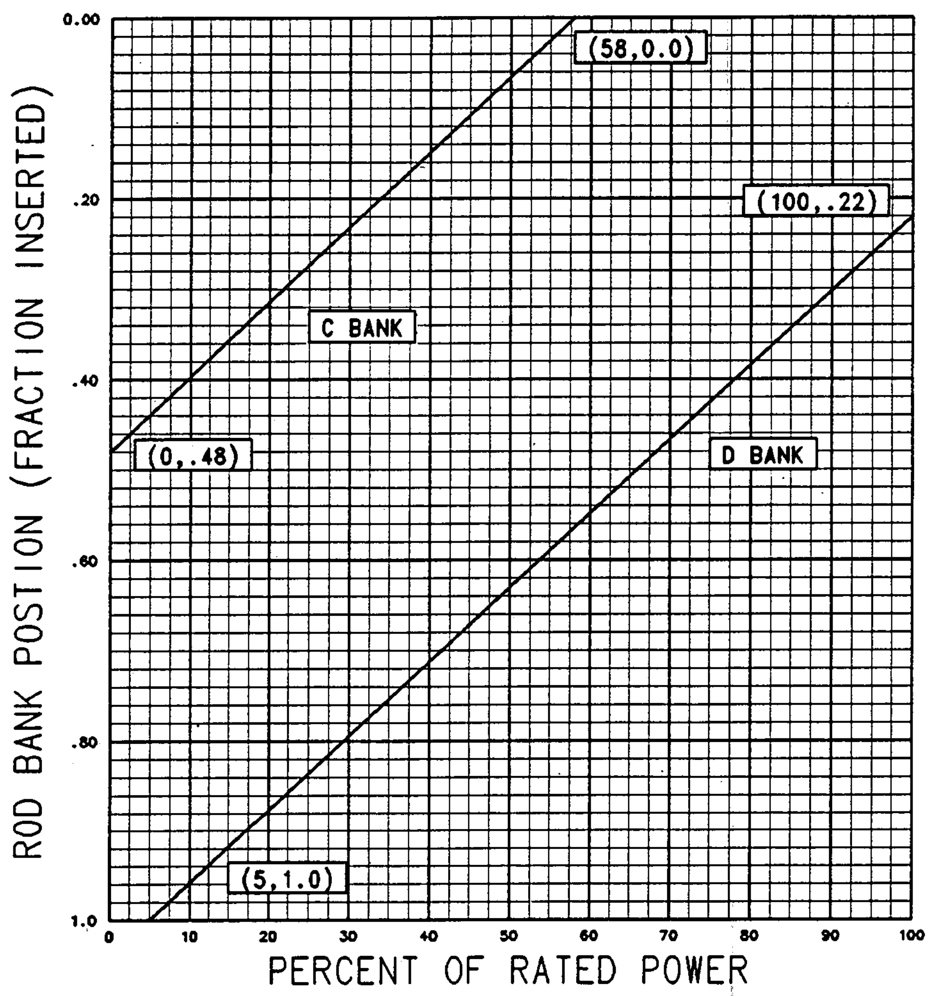

FIGURE 3

CONTROL ROD INSERTION LIMITS AS A FUNCTION OF POWER