RELOAD SAFETY EVALUATION

INDIAN POINT NUCLEAR PLANT
UNIT 2, CYCLE 2

January 1976

Edited by
J. Skaritka

Work Performed Under Shop Order IPBF-460

Approved: A. L. Field Jr. for R. J. W.

R. J. Wiggins, Manager
Fuel Licensing and Coordination
Nuclear Fuel Division

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION AND SUMMARY

The Indian Point Nuclear Power Plant, Unit Number 2, Cycle 1 achieved initial criticality in May 1973. The plant is being operated at a maximum 100% rated power of 2758 MWt through the end of Cycle 1 (estimated at April 15, 1976).

This report presents an evaluation for Cycle 2 and 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. Rather, heavy dependence has been placed on the analysis presented in the FSAR[1] and the Fuel Densification Report[2]. Fuel performance and FSAR design basis accidents have been shown to be acceptable by demonstrating that Cycle 2 results satisfy the design and safety limits for Cycle 1.

For Cycle 2 operation Indian Point Unit 2 will replace 65 Region 1 and seven Region 2 assemblies with 72 Region 4 assemblies. The core loading pattern for Cycle 2 is shown in Figure 1. All of the accidents analyzed and reported in the FSAR which could potentially be affected by fuel reload have been reviewed for the Cycle 2 design described herein. The results for those requiring reanalysis have been included and the justification for the applicability of previous results for the remainder is presented. This conclusion is based on the assumption that: (1) Cycle 1 operation is terminated after 16,000 $\pm$ 1000 MWD/MTU and (2) there is adherence to plant operating limitations discussed later in required modifications to the Technical Specifications (Section 4.0 and Reference 6).

Nominal design parameters for Cycle 2 are 2758 MWt core power, 2250 psia system pressure, and core average linear power of 5.8 kw/ft.

2.0 REACTOR DESIGN

2.1 Mechanical Design

The mechanical design of Region 4 fuel is identical to Regions 2 and 3 fuel except as noted in Table 1. Clad flattening will not occur during Cycle 2. The maximum irradiation time for the Region 2 and 3 assemblies from Cycle 1 is 12,800 EFPH. Maximum expected additional irradiation during Cycle 2 is 8700 EFPH. This gives a total of 21,500 EFPH. Clad flattening time is predicted to be greater than 30,000 EFPH for Regions 2 and 3, using the current Westinghouse evaluation model[3].

2.2 Nuclear Design

Cycle 2 core loading is designed to meet an $F_Q^T \times P$ limit of ≤ 2.32. The normalized limiting F_Q^T as a function of core height, which satisfies the ECCS FAC criteria, is shown in Figure 2.

Table 2 provides a comparison of the Cycle 2 kinetics characteristics with the current limit based on previously submitted accident analysis.

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 analysis. The available shutdown margin exceeds the minimum required. Figures 3 and 4 give the control rod insertion limits to assure that peaking factors are not exceeded during anticipated power control maneuvers.

It is recognized that the Nuclear Regulatory Commission is considering LOCA power spikes and DNBR penalties to accommodate fuel rod bow effects which have been observed in a number of nuclear reactors. The Indian Point 2 reactor utilizes a high-parasitic nine grid fuel assembly skeleton, for which rod bowing has not been observed in nuclear plants using the same skeletons. For example, the Ginna reactor is in its fifth cycle of operation and has not observed any significant rod bowing in recent fuel cycles. Therefore, it is not necessary to consider any postulated rod bow effects or penalties.

2.3 Thermal and Hydraulic Design

No significant variations in thermal margins will result from the Cycle 2 reload. The present DNB core limits have been found to be conservative.

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] and fuel densification reports[2], 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 2. The time dependent densification model[4] was used for fuel temperature evaluations. F_Q is maintained at or below 2.32, according to the normalized F_Q envelope in Figure 2. This curve is satisfied by the power control maneuvers allowed by the Technical Specification, which assure that the FAC criteria are met for a spectrum of small and large LOCAs. In order to accommodate an increase in the K(Z) third line coordinate (Figure 2) from (12.0, 0.43) to (12.0, 0.54) for Cycle 2, the small break LOCA was reanalyzed and was found to satisfy the FAC criteria.

3.2 Accident Evaluation

The effects of the core reload on the design basis and postulated incidents analyzed in the FSAR have been examined. In most cases, it was found that the effects can be accomodated 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. Therefore, the conclusions presented in the FSAR and densification reports are still valid.

A core reload can typically affect accident analysis input parameters in three major areas: core kinetics 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.

Kinetics Parameters - A comparison of Cycle 2 kinetics parameters with the current limits is given in Table 2. The moderator temperature coefficient and prompt neutron lifetime are within bounds of the current limit. Table 2 shows that the Cycle 2 Doppler coefficient exceeds the Cycle 1 values. Accidents potentially affected are loss of flow, locked rotor, and loss of load. Sensitivity studies indicate that the loss of flow and locked rotor transient results, in terms of minimum DNB ratio, are relativelv insensitive to the Doppler coefficient assumed. Thus, only the loss of load transient has been reanalyzed due to the higher (in absolute value) Doppler coefficient. Results of this analysis are discussed in Section 3.3.

Table 4 shows that the Cycle 2 beginning of life delayed neutron fraction for rod ejection purposes is less than the Cycle 1 values. The beginning-of-life rod ejection transient cases were therefore reanalyzed using the reduced delayed neutron fraction. Results of this analysis are discussed in Section 3.3.

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 or equal to the current limit. Cycle 2 ejected rod worths are shown in Table 4. As shown in the table, ejected rod worths for Cycle 2 are within the bounds of the current limits.

The trip reactivity insertion rate for Cycle 2 has been calculated to be slightly more rapid in the upper third of the core than for Cycle 1, but slower for the remainder of the core. The total trip reactivity is unchanged. The effects of this altered trip reactivity insertion rate curve have been evaluated for those accidents affected and compared with Cycle 1 analyses. Slow transients are relatively insensitive to changes in trip reactivity insertion rate, and therefore need not be reanalyzed due to the change in trip reactivity versus rod position. Fast transients such as rod ejection and rod withdrawal from subcritical, in which negative reactivity insertion is due primarily to Doppler feedback, will be unaffected by the change in trip reactivity insertion rate since the transient is essentially turned around by Doppler feedback before rod insertion starts. The effect of variations in trip reactivity insertion rates for the rod withdrawal at power incidents has also been evaluated. Since the minimum DNB ratio for the transient occurs at relatively low reactivity insertion rates, the value of the minimum DNB ratio is unaffected.

The loss of flow transient is sensitive to the rate of trip reactivity insertion. Since the calculated Cycle 2 insertion rate is more rapid in the upper third of the core, core power will be reduced earlier in the transient than for Cycle 1. For this reason, the loss of flow accident has not been reanalyzed, and the conclusions reached in the previous applicable safety analysis are still valid.

Core Peaking Factors - Evaluation of peaking factors for the rod out of position and dropped RCCA incidents shows that DNBR is maintained above 1.3. For the dropped bank incident, the minimum DNBR criteria of 1.3 is satisfied without taking credit for a turbine runback. Table 4 shows the peaking factors following control rod ejection. The peaking factors are less than the current limit values for all cases except EOL - HZP. Thus the EOL - HZP rod ejection transient case has been reanalyzed using the higher peaking factor. A peaking factor evaluation for the hvpothetical steambreak transient showed that, using the new fuel densification model[4], the DNBR design criterion given in Section 14.2.5 of the FSAR was met for all cases.

3.3 Incidents Reanalyzed

The loss of load transient was reanalyzed using the Cycle 2 Doppler coefficient. The analysis was performed using the methods and assumptions employed for Cycle 1[1]. The analysis shows that the minimum DNB ratio is greater than 1.3, and that the pressurizer safety valves and steam generator safety valves are more than adequate to limit the maximum pressures in the reactor coolant system and the main steam system respectively to acceptable values. Thus, the conclusions presented in Reference 1 remain valid.

The rod ejection transient was also reanalyzed. Beginning-of-life cases were reanalyzed due to a lower Cycle 2 delayed neutron fraction (see Table 4). The EOL - HZP case was reanalyzed due to a higher peaking factor. Lastly, the EOL - HFP case was reanalyzed since the average fuel temperature conservatively assumed at the initial hot spot linear power density exceeded that previously used for this case* by approximately 295°F. The effect of the higher initial fuel temperature is to increase the peak transient fuel and clad temperature following the rod ejection. All cases were analyzed using the methods described for Cycle 1[2]. The results, presented in Table 5, show that the fuel rod at the hot spot does not exceed the limiting criteria[5].

Due to the revised third line segment resulting from a higher F_Q in the upper core (See Figure 2), the small break LOCA accident was reanalyzed. Results show that the ECCS FAC is satisfied up to full power conditions.

---

*This fuel temperature was used only for the EOL ejected rod analysis. Fuel temperatures previously used in other incidents are unaffected.

4.0 TECHNICAL SPECIFICATION

This section contains the technical content of proposed changes to the Indian Point Unit 2 Technical Specifications. These changes are consistent with the plant operation necessary for the design and safety evaluation conclusions stated previously to remain valid.

4.1 Specification 3.10.1 - Control Rod Insertion Limits

Revision: Replace Figures 3.10-1 & 3.10-2 of the existing Technical Specifications and Figures 3.10-3 and 3.10-4 of the Technical Specifications proposed on July 9, 1975 (6) and 3.10-4 and 3.10-5 of the Technical Specifications proposed on September 6, 1974 (6) with the attached Figures 3 and 4.

This assures that core peaking factor limits are not exceeded during power control maneuvers allowed by the Technical Specifications.

Bases: In the course of design a set of insertion limits are selected which is estimated to meet the insertion limit criteria. The following criteria are checked in the design process:

1. The shutdown margin is maintained by calculating the inserted reactivity (reactivity allowance) for the control rods at the insertion limit.
2. For rod positions allowed in normal operation, the enthalpy rise hot channel factor, $F_{\Delta H}$, must be maintained within limits.
3. The consequences of an ejected control rod assembly from the allowed insertion must be within the accepted limits.
4. Statically misaligning a control assembly will not violate the thermal design basis with respect to DNBR.

If any of the above are not met the insertion limit must be adjusted accordingly. The design requirements for Cycle 2 are met by confirming that the above criteria are satisfied for the Cycle 2 insertion limits.

4.2 Specification 3.10 - Control Rod and Power Distribution Limits

Replace Figure 3.10-2 in Section 3.10 of the Consolidated Edison proposed revisions[6] to the Technical Specifications with the enclosed Figure 2. The increase in the K(Z) third line coordinate in Figure 2 from (12.0, 0.431) to (12.0, 0.54) assures that the Cycle 2 power control maneuvers allowed by the Technical Specifications will be satisfied. For this modified third line K(Z) segment, the small break LOCA was reanalyzed and was found to satisfy the FAC criteria.

The reference transmittal[6] specifies a change in the design basis hot channel factor as:

$$F_{\Delta H} \leq 1.55\ [1 + 0.2\ (1-P)]$$

This is a basis for the Cycle 2 nuclear design.

5.0 REFERENCES

1. Final Safety Analysis Report - Indian Point Unit Number 2, Docket Number 50-247

2. Fuel Densification - Indian Point Nuclear Generating Station Unit Number 2, January 1973; and Addendum dated March 22, 1973

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

4. Hellman, J. M. (Ed.), "Fuel Densification Experimental Results and Model for Reactor Application", WCAP 8218-P-A, March 1975 (Proprietary) and WCAP 8219-A, March 1975 (Non Proprietary

5. Risher, D. H. Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods", WCAP 7588, Revision 1, December 1971. (Non Proprietary)

6. Consolidated Edison Applications for Amendments to the Technical Specifications Filed with the Nuclear Regulatory Commission, dated September 6, 1974 and July 9, 1975; Indian Point Unit No. 2, Docket No. 50 - 247.

Table 1

FUEL ASSEMBLY DESIGN PARAMETERS

INDIAN POINT UNIT NUMBER 2 - CYCLE 2

Region	2	3	4
Enrichment (w/o of U 235)*	2.80	3.30	3.10
Density (percent theoretical)*	94.5	94.3	95.0
Number of Assemblies	57	64	72
Approximate Burnup at Beginning of Cycle 2 (MWD/MTU)	17800	12700	0

*Regions 2 and 3 are as-built values. Region 4 values are design. However, Region 4 used an average density of 94.5% theoretical for thermal evaluations.

Table 2

KINETICS CHARACTERISTICS

INDIAN POINT UNIT NUMBER 2 - CYCLE 2

	Cycle 1	Cycle 2
Moderator Temperature Coefficient, $(\Delta\rho/°F)x10^4$	-3.5 to 0.0	-3.5 to 0.0*
Doppler Coefficient $(\Delta\rho/°F)x10^5$	-1.8 to -1.1	-2.0 to -1.1
Delayed Neutron Fraction β_{eff} (percent)	.50 to .70	.50 to .70
Prompt Neutron Lifetime (μ sec)	18	18
Maximum Differential Rod Worth of Two Banks Moving Together at HZP $(\Delta\rho/sec)x10^5$	80.0	$\leq$80.0

*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 2 – CYCLE 2

	Cycle 1 (2) EOC	Cycle 2 BOC	Cycle 2 EOC
Control Rod Worth (percent $\Delta\rho$)			
All Rods Inserted Less Worst Stuck Rod	7.28	5.85	6.17
(1) Less 10%	6.55	5.26	5.55
Control Rod Requirements (percent $\Delta\rho$)			
Reactivity Defects (Doppler, Tavg, Void, Redistribution)	3.03	1.62	2.61
Rod Insertion Allowance	.70	.50	.50
(2) Total Requirements	3.73	2.12	3.11
Shutdown Margin [(1)-(2)] (percent $\Delta\rho$)	2.82	3.14	2.44
Required Shutdown Margin (percent $\Delta\rho$)	1.95	1.0	1.95

Table 4

ROD EJECTION PARAMETERS

INDIAN POINT UNIT 2 CYCLE 2

	Previous Analysis Values	Cycle 2	Values Used In Reanalysis
HZP-BOL			
Max. Ejected Rod Worth, %$\Delta\rho$	0.74	$\leq$0.74	0.74
Max. F_Q^N	15.3	$\leq$15.3	15.3
β_{eff}	0.007	0.0058	0.0058
HFP-BOL			
Max. Ejected Rod Worth, %$\Delta\rho$	0.27	$\leq$0.27	0.27
Max. F_Q^N	5.71	$\leq$5.71	5.71
β_{eff}	0.007	0.0058	0.0058
HZP-EOL			
Max. Ejected Rod Worth, %$\Delta\rho$	0.67	0.52	0.67
Max. F_Q^N	14.9	15.3	15.3
β_{eff}	0.005	0.005	0.005
HFP-EOL			
Max. Ejected Rod Worth, %$\Delta\rho$	0.23	$\leq$0.23	0.23
Max. F_Q^N	4.84	$\leq$4.84	4.84
β_{eff}	0.005	0.005	0.005
Initial average fuel temperature, °F	2170	2465	2465

TABLE 5

RESULTS OF ROD EJECTION ANALYSIS HOT SPOT FUEL AND CLAD TEMPERATURES

INDIAN POINT UNIT 2 CYCLE 2

	BOL	BOL	EOL	EOL
Initial Power,%	0	102	0	102
Maximum Fuel Pellet Center Temperature (°F)	2932	5141	2751	4638
Maximum Fuel Average Temperature (°F)	2557	4037	2382	3324
Maximum Clad Temperature (°F)	1955	2308	1834	1771
Maximum Fuel Enthalpy (cal./gm)	103	176	95	140

Figure 1

CORE LOADING PATTERN
INDIAN POINT UNIT 2 - CYCLE 2

90°

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

0° (left) — 180° (right, at row J)

270°

x y	- Fuel Region - BPs from Cycle 1
SS	- Secondary Source Assembly

Figure 2

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE

INDIAN POINT UNIT 2 CYCLE 2

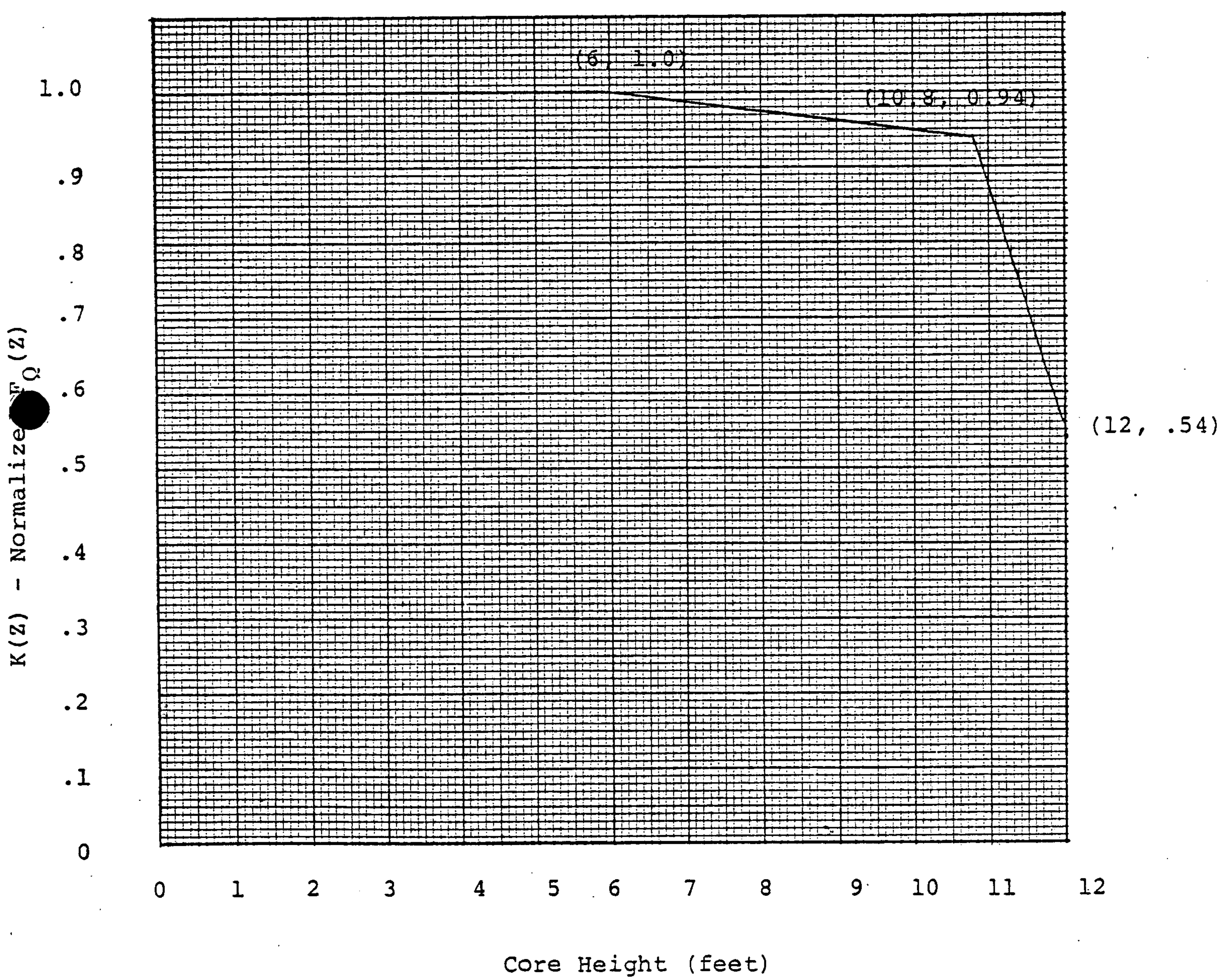

Figure 3

ROD BANK INSERTION LIMITS
(Four Loop Operation)

INDIAN POINT UNIT 2

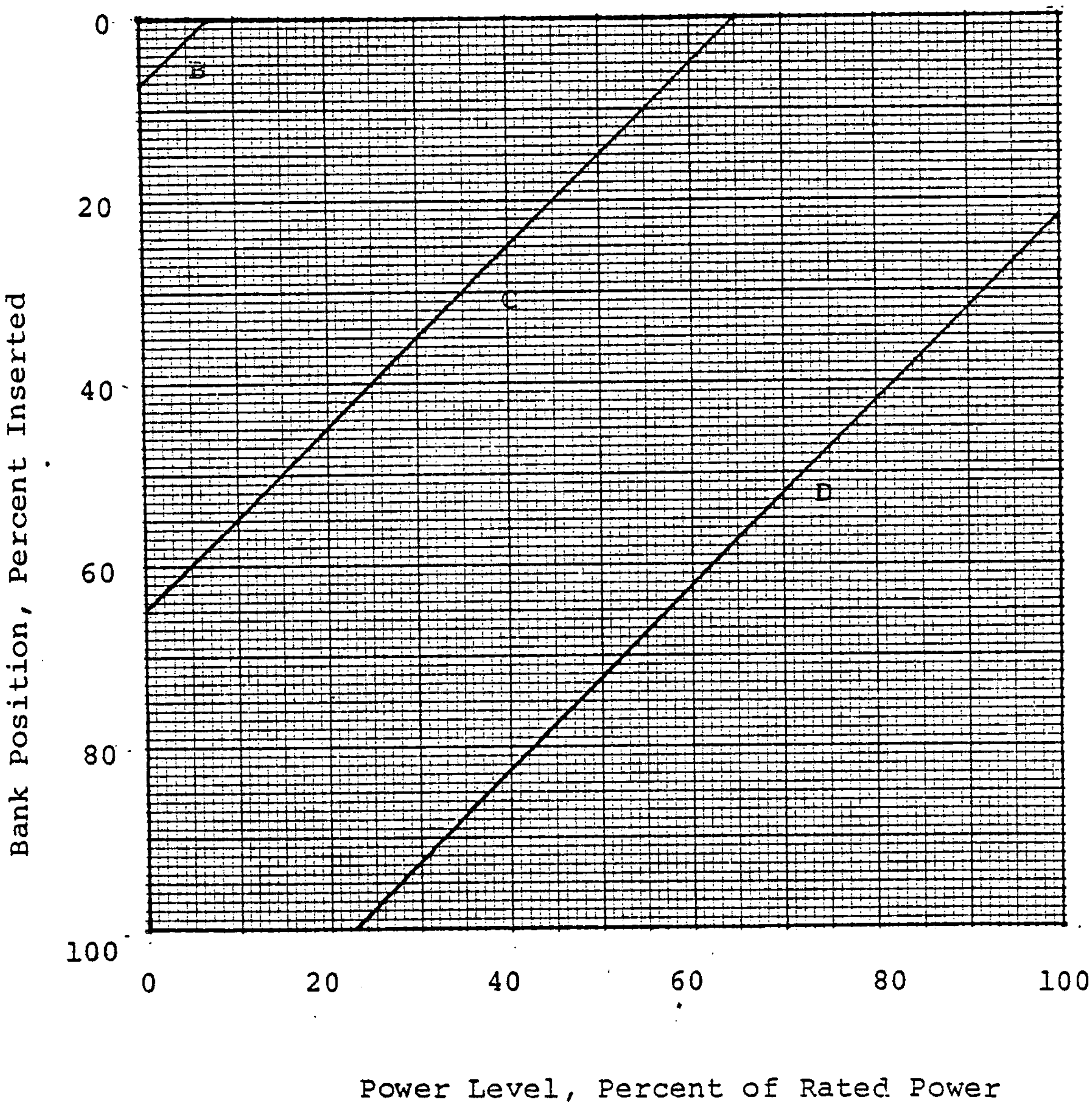

Figure 4

ROD BANK INSERTION LIMITS
(Three Loop Operation)

INDIAN POINT UNIT NO. 2