RELOAD SAFETY EVALUATION
INDIAN POINT NUCLEAR PLANT
UNIT 3, CYCLE 4

FEBRUARY 1982

Edited by
J. Skaritka

Approved: M. G. Arlotti

M. G. Arlotti, Manager
Fuel Licensing & Coordination
Nuclear Fuel Division

8306140598 820228
PDR ADOCK 05000286
P PDR

A001
1/1

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 is in its third cycle of operation. The unit is expected to be refueled and ready for Cycle 4 startup in April 1982.

This report presents an evaluation for Unit 3 Cycle 4, 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, "Westinghouse Reload Safety Evaluation Methodology"[1].

Based upon the above referenced methodology, only those incidents analyzed and reported in the FSAR[2] which could potentially be affected by the fuel reload have been reviewed for the Cycle 4 design described herein. No new analyses were required for the Cycle 4 design. The justification for the applicability of previous results is provided. The NRC has reviewed safety analyses and has approved technical specification changes[3] submitted during Cycle 3 to allow a reduction in Thermal Design Flow (TDF) and to account for up to a maximum 12% steam generator tube plugging.

1.2 GENERAL DESCRIPTION

The Indian Point 3 Cycle 4 reactor core is comprised of 193 fuel assemblies arranged in the core loading pattern configuration shown in Figure 1. During the Cycle 3/4 refueling, 76 fuel assemblies will be replaced with Regions 6-1 and 6-2 fresh fuel. 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)	3025 (100% rated)
System Pressure (psia)	2250
Core Inlet Temperature (°F)	541.2*
Thermal Design Flow (gpm)	342,400*
Average Linear Power Density (kw/ft) (based on best estimate hot, densified core average stackheight of 143.9 inches)	6.24

1.3 CONCLUSIONS

From the evaluation presented in this report, it is concluded that the Cycle 4 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 3 burnup between 11,700 and 13,000 MWD/MTU.

2. Cycle 4 burnup will not exceed 14,500 MWD/MTU, which includes a power/temperature coastdown.

3. There is adherance to plant operating limitations as given in the Technical Specifications.

*Accounts for up to 12% steam generator tube plugging and TDF represents 95.4% BOL Cycle 3 TDF.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The mechanical design of the Region 6-1 and 6-2 fuel assemblies is the same as the Region 5 assemblies. Table 1 compares pertinent design parameters of the various fuel regions. The Regions 6-1 and 6-2 fuel has been designed according to the fuel performance model in Reference 4. The fuel is designed and operated so that clad flattening will not occur, as predicted by the Westinghouse model[5].

Westinghouse 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

The Cycle 4 core loading satisfies the F_Q^T x P ECCS limit of $\leq$ 2.04 x K(z), given in Figure 2, which is necessary to meet the current PCT required by the NRC. The control rod insertion limits are unchanged from Cycle 3 limits. In addition, the flux difference (ΔI) bandwidth during normal operation is unchanged from Cycle 3 ±5% ΔI.

Table 2 provides a comparison of the cycle 4 kinetics characteristics with the current analysis value based on previously submitted accident analysis. It can be seen from the table that except for the Doppler Temperature Coefficient, the Cycle 4 values fall within the range of the previous analysis value. 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 allowance at BOL is the as-calculated value.

Thirty-two Region 6-1 and twenty-four Region 6-2 fuel assemblies contain fresh or depleted burnable absorber rods. Two sets of eight Region 5 fuel assemblies also contain fresh or depleted burnable absorber rods. Two other Region 5 fuel assemblies contain secondary source rod assemblies. See Figure 1 for the location of burnable absorber and source rods.

Provisions have been made to accommodate depleted burnable absorber clusters either with or without the wet annular Al_2O_3-B_4C burnable absorber demonstration rodlets. This enables the further irradiation of depleted burnable absorber clusters which contain some wet annular burnable absorber rodlets. The replacement of the standard depleted burnable absorber clusters with the demonstration clusters will have negligible nuclear effects on the power distribution and operation of the plant. As shown in Addendum 1 to the Cycle 3 RSE[7], no safety limits will be exceeded in the unlikely event of the burnable absorber demonstration rodlets failing and the B_4C absorber material being lost from the rodlets.

2.3 THERMAL AND HYDRAULIC DESIGN

No significant variations in thermal margins will result from the Cycle 4 reload. The DNB core limits, contained in the recent technical specification changes[3], are based on the conditions given in Section 1.2.

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 and coastdown during Cycle 4. For the overpower transient, the fuel centerline temperature limit of 4700^{o}F can be accommodated with margin during Cycle 4. The time dependent densification model[8] was used for fuel temperature evaluations. The LOCA limit is met by maintaining F_Q x P at or below 2.04 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[1], fuel densification report[9], and Cycle 3 were 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. Therefore, the conclusions presented in the FSAR and subsequent analyses are still valid.

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 4 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 4 core physical parameters with the previous cycle parameters is presented in Table 2. All parameters in Table 2, except the Doppler temperature coefficient, were found to be within the range of values used in previous safety analyses. The most negative Doppler temperature coefficient is -1.9 pcm/^{o}F compared to the previous value of -1.60 pcm^{o}/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 drops. For the most severe reactivity addition accident (startup of an inactive loop), this amounts to less than a 2% increase in total positive reactivity insertion. This would yield a negligible increase in peak power which can be accommodated in all of the FSAR cooldown accidents. In addition, the Doppler power coefficient actually calculated for this reload is larger than that assumed in the FSAR, and this would more than compensate for the Doppler temperature coefficient changes. Thus, no reanalysis is required. 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 4 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 4 does not exceed the current limit. Cycle 4 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.

The steamline break transients (FSAR Section 14.2.5) were evaluated for Cycle 4 using the same methods as the Cycle 3 reanalysis. The evaluations showed that the Cycle 4 transients are within the bounds of the Cycle 3 analysis.

4.0 REFERENCES

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

2. Final Safety Analysis Report - Indian Point Unit Number 3, Docket Number 50-286.

3. Letter from NRC (J.O. Thoma) to PASNY (G.T. Berry), Docket No. 50-286, Subject: Amendment 40 to Indian Point Unit 3 Operating License DPR-64; November 13, 1981.

4. Miller, J.V., (ED), "Improved Analytical Model Used in Westinghouse Fuel Rod Design Computations," WCAP-8785, October 1976.

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

6. Skaritka, J., Iorii, J.A., "Operational Experience with Westinghouse Cores," WCAP-8183, Revision 10, May 1981.

7. Skaritka, J., Editor, "Reload Safety Evaluation-Indian Point Unit 3, Cycle 3," August 1979.

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

9. Fuel Densification-Indian Point Nuclear Generating Station Unit Number 3, WCAP-8146 (Proprietary) and WCAP-8147 (Non-Proprietary), July 1973.

TABLE 1

FUEL ASSEMBLY DESIGN PARAMETERS
INDIAN POINT UNIT NUMBER 3 - CYCLE 4

Region	4	5	6-1	6-2
Enrichment (w/o of U 235)*	3.10	3.30	3.20	3.40
Density (percent theoretical)*	94.7	94.7	94.5	94.5
Number of Assemblies	41	76	36	40
Approximate Burnup at Beginning of Cycle 4 (MWD/MTU)+	22,100	12,100	0	0

*All fuel region enrichments except regions 6-1 and 6-2 are as-built values. An average density of 94.5% theoretical was used for Region 6-1 and 6-2 design evaluations.

+Based on a Cycle 3 core average burnup of 12,000 MWD/MTU.

1512F:6

TABLE 2

KINETICS CHARACTERISTICS

INDIAN POINT UNIT NUMBER 3 - CYCLE 4

	Previous Analysis Values (2), (7), (9),	Cycle 4
Moderator Temperature Coefficient, (PCM/^{o}F)*	-35 to 0.0	-35 to 0.0**
Least Negative Doppler - Only Power Coefficient, Zero to Full Power (pcm/% power)*	-7.0 (constant)	-7.0 (constant)
Most Negative Doppler - Only Power Coefficient Zero to Full Power (pcm/% power)*	-27.7 to -27.1	-27.7 to -27.1
Delayed Neutron Fraction β_{eff} (percent	.44 to .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/^{o}F)	-1.6 to -1.1	-1.9 to -1.1

* 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.

1512F:6

TABLE 3

SHUTDOWN REQUIREMENTS AND MARGINS
INDIAN POINT UNIT NUMBER 3 - CYCLES 3 AND 4

	Cycle 3		Cycle 4	
	BOC	EOC	BOC	EOC
Control Rod Worth (percent $\Delta\rho$)				
All Rods Inserted Less Worst Stuck Rod	5.09	5.99	5.96	6.66
(A) Less 10%	4.58	5.39	5.36	5.99
Control Rod Requirements (percent $\Delta\rho$)				
Reactivity Defects (Doppler, Tavg, Void, Redistribution)	1.80	2.90	1.75	2.70
Rod Insertion Allowance	0.26	0.55	.50	.60
(B) Total Requirements	2.06	3.45	2.25	3.30
Shutdown Margin [(A)-(B)] (percent $\Delta\rho$)	2.52	1.94	3.11	2.69
Required Shutdown Margin (percent $\Delta\rho$)(1)	1.0	1.72	1.0	1.72

1512F:6

FIGURE 1

CORE LOADING PATTERN
INDIAN POINT UNIT 3 - CYCLE 4

90°

0° | 180°

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

270°

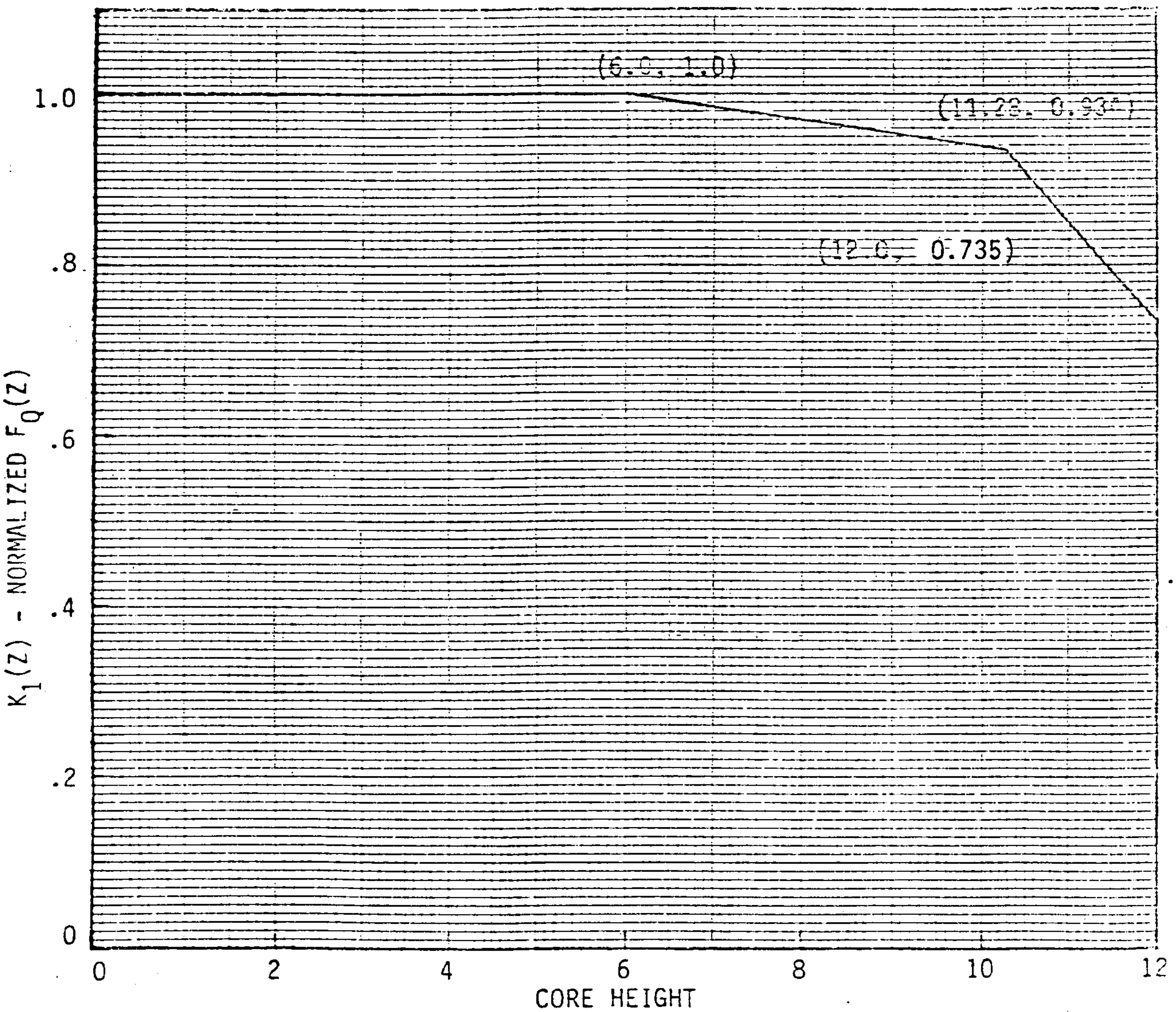
X 12 SS	
X	- Region Number
12	- Number of new burnable absorber rods
SS	- Secondary Source Rods

* 12D	
*	- Locations for possible insertion of burnable absorber demonstration rodlets (see Section 2.2)
12D	- 12 Depleted burnable absorber rods from Cycle 3

FIGURE 2

HEAT FLUX HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE
FOUR-LOOP OPERATION

BASIS: $F_Q(Z) \times P$ ECCS LIMIT OF 2.04



RELOAD SAFETY EVALUATION
INDIAN POINT NUCLEAR PLANT
UNIT 3, CYCLE 4
REVISION 1

MARCH 1982

Edited by
J. Skaritka

Approved: M G Arlotti

M. G. Arlotti, Manager
Fuel Licensing & Coordination
Nuclear Fuel Division

1855F:6

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 is in its third cycle of operation. The unit is expected to be refueled and ready for Cycle 4 startup in April 1982.

This report presents an evaluation for Unit 3 Cycle 4, 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, "Westinghouse Reload Safety Evaluation Methodology"[1].

Based upon the above referenced methodology, only those incidents analyzed and reported in the FSAR[2] which could potentially be affected by the fuel reload have been reviewed for the Cycle 4 design described herein. No new analyses were required for the Cycle 4 design. The justification for the applicability of previous results is provided. Previous results include safety analyses[3][4] and proposed technical specification changes[5] submitted during Cycle 3 to allow a reduction in Thermal Design Flow (TDF) and to account for up to a maximum 24% steam generator tube plugging.

This report represents a revision to the original February 1982 Cycle 4 RSE which evaluated 100% rated power at 12% steam generator tube plugging. Changes from the February 1982 RSE are noted by bars in the margins.

1.2 GENERAL DESCRIPTION

The Indian Point 3 Cycle 4 reactor core is comprised of 193 fuel assemblies arranged in the core loading pattern configuration shown in Figure 1. During the Cycle 3/4 refueling, 76 fuel assemblies will be replaced with Regions 6-1 and 6-2 fresh fuel. 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)	3025 (100% rated)
System Pressure (psia)	2250
Core Inlet Temperature (^{o}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

1.3 CONCLUSIONS

From the evaluation presented in this report, it is concluded that the Cycle 4 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 3 burnup between 11,700 and 13,000 MWD/MTU.

2. Cycle 4 burnup will not exceed 14,500 MWD/MTU, which includes a power/temperature coastdown.

3. There is adherance to plant operating limitations as given in the Technical Specifications and the proposed Technical Specifications(5) for a reduced TDF.

*Accounts for up to 24% steam generator tube plugging and TDF represents 90.2% BOL Cycle 3 TDF.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The mechanical design of the Region 6-1 and 6-2 fuel assemblies is the same as the Region 5 assemblies. Table 1 compares pertinent design parameters of the various fuel regions. The Regions 6-1 and 6-2 fuel has been designed according to the fuel performance model in Reference 6. The fuel is designed and operated so that clad flattening will not occur, as predicted by the Westinghouse model[7].

Westinghouse has had considerable experience with Zircaloy clad fuel. This experience is extensively described in WCAP-8183, "Operational Experience with Westinghouse Cores[8]." This report is updated annually.

2.2 NUCLEAR DESIGN

The Cycle 4 core loading satisfies the $F_Q^T \times P$ ECCS limit of $\leq 2.2 \times K(z)$, given in Figure 2, which is necessary to meet the current PCT required by the NRC. The control rod insertion limits are unchanged from Cycle 3 limits. In addition, the flux difference (ΔI) bandwidth during normal operation is unchanged from Cycle 3 $\pm 5\%$ ΔI.

Table 2 provides a comparison of the cycle 4 kinetics characteristics with the current analysis value based on previously submitted accident analysis. It can be seen from the table that except for the Doppler Temperature Coefficient, the Cycle 4 values fall within the range of the previous analysis value. 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 allowance at BOL is the as-calculated value.

Thirty-two Region 6-1 and twenty-four Region 6-2 fuel assemblies contain fresh or depleted burnable absorber rods. Two sets of eight Region 5 fuel assemblies also contain fresh or depleted burnable absorber rods. Two other Region 5 fuel assemblies contain secondary source rod assemblies. See Figure 1 for the location of burnable absorber and source rods.

Provisions have been made to accommodate depleted burnable absorber clusters either with or without the wet annular Al_2O_3-B_4C burnable absorber demonstration rodlets. This enables the further irradiation of depleted burnable absorber clusters which contain some wet annular burnable absorber rodlets. The replacement of the standard depleted burnable absorber clusters with the demonstration clusters will have negligible nuclear effects on the power distribution and operation of the plant. As shown in Addendum 1 to the Cycle 3 RSE(9), no safety limits will be exceeded in the unlikely event of the burnable absorber demonstration rodlets failing and the B_4C absorber material being lost from the rodlets.

2.3 THERMAL AND HYDRAULIC DESIGN

No significant variations in thermal margins will result from the Cycle 4 reload. The DNB core limits, which are given in the proposed technical specification(5), are based on the conditions given in Section 1.2.

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 and coast-down during Cycle 4. For the overpower transient, the fuel centerline temperature limit of $4700^{\circ}F$ can be accommodated with margin during Cycle 4. 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 2.2 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[1], fuel densification report[11], and Cycle 3[3][4][9] were 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. Therefore, the conclusions presented in the FSAR and subsequent analyses are still valid.

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 4 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 4 core physical parameters with the previous cycle parameters is presented in Table 2. All parameters in Table 2, except the Doppler temperature coefficient, were found to be within the range of values used in previous safety analyses. The most negative Doppler temperature coefficient is -1.9 pcm/^{O}F compared to the previous value of -1.60 pcm^{O}/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 drops. For the most severe reactivity addition accident (startup of an inactive loop), this amounts to less than a 2% increase in total positive reactivity insertion. This would yield a negligible increase in peak power which can be accommodated in all of the FSAR cooldown accidents. In addition, the Doppler power coefficient actually calculated for this reload is larger than that assumed in the FSAR, and this would more than compensate for the Doppler temperature coefficient changes. Thus, no reanalysis is required. 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 4 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 4 does not exceed the current limit. Cycle 4 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.

The steamline break transients (FSAR Section 14.2.5) were evaluated for Cycle 4 using the same methods as the Cycle 3 reanalysis. The evaluations showed that the Cycle 4 transients are within the bounds of the Cycle 3 analysis.

4.0 REFERENCES

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

2. Final Safety Analysis Report - Indian Point Unit Number 3, Docket Number 50-286.

3. Letter from Westinghouse (J. D. Campbell) to PASNY (Jim Clabby), INT-81-557, Subject: 24% Tube Plugging Safety Analysis (Non-LOCA), November 13, 1981.

4. Letter from Westinghouse (J. D. Campbell) to PASNY (Jim Clabby) INT-81-560, Subject: 24% Tube Plugging Safety Analysis (LOCA),, November 17, 1981.

5. Letter from Westinghouse (F. Noon) to Pasny (Jim Clabby), INT-81-551, Subject: 24% Tube Plugging Technical Specification Changes (Non-LOCA), November 4, 1981.

6. Miller, J.V., (ED), "Improved Analytical Model Used in Westinghouse Fuel Rod Design Computations," WCAP-8785, October 1976.

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

8. Skaritka, J., Iorii, J.A., "Operational Experience with Westinghouse Cores," WCAP-8183, Revision 10, May 1981.

9. Skaritka, J., Editor, "Reload Safety Evaluation-Indian Point Unit 3, Cycle 3," August 1979.

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

11. Fuel Densification-Indian Point Nuclear Generating Station Unit Number 3, WCAP-8146 (Proprietary) and WCAP-8147 (Non-Proprietary), July 1973.

TABLE 1

FUEL ASSEMBLY DESIGN PARAMETERS
INDIAN POINT UNIT NUMBER 3 - CYCLE 4

Region	4	5	6-1	6-2
Enrichment (w/o of U 235)*	3.10	3.30	3.20	3.40
Density (percent theoretical)*	94.7	94.7	94.5	94.5
Number of Assemblies	41	76	36	40
Approximate Burnup at Beginning of Cycle 4 (MWD/MTU)+	22,100	12,100	0	0

*All fuel region enrichments except regions 6-1 and 6-2 are as-built values. An average density of 94.5% theoretical was used for Region 6-1 and 6-2 design evaluations.

+Based on a Cycle 3 core average burnup of 12,000 MWD/MTU.

TABLE 2

KINETICS CHARACTERISTICS

INDIAN POINT UNIT NUMBER 3 - CYCLE 4

	Previous Analysis Values (2), (3), (9), (11)	Cycle 4
Moderator Temperature Coefficient, (PCM/^{o}F)*	-35 to 0.0	-35 to 0.0**
Least Negative Doppler - Only Power Coefficient, Zero to Full Power (pcm/% power)*	-7.0 (constant)	-7.0 (constant)
Most Negative Doppler - Only Power Coefficient Zero to Full Power (pcm/% power)*	-27.7 to -27.1	-27.7 to -27.1
Delayed Neutron Fraction β_{eff} (percent	.44 to .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/^{o}F)	-1.6 to -1.1	-1.9 to -1.1

* 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 3 AND 4

	Cycle 3		Cycle 4*	
	BOC	EOC	BOC	EOC
Control Rod Worth (percent $\Delta\rho$)				
All Rods Inserted Less Worst Stuck Rod	5.09	5.99	5.96	6.66
(A) Less 10%	4.58	5.39	5.36	5.99
Control Rod Requirements (percent $\Delta\rho$)				
Reactivity Defects (Doppler, Tavg, Void, Redistribution)	1.80	2.90	1.81	2.84
Rod Insertion Allowance	0.26	0.55	.50	.60
(B) Total Requirements	2.06	3.45	2.31	3.44
Shutdown Margin [(A)-(B)] (percent $\Delta\rho$)	2.52	1.94	3.05	2.55
Required Shutdown Margin (percent $\Delta\rho$)(1)	1.0	1.72	1.0	1.72

*The Total Requirements (β) value have been changed to accommodate 24% steam generator tube plugging with a core T_{avg} = 577.6°F.

FIGURE 1

CORE LOADING PATTERN
INDIAN POINT UNIT 3 - CYCLE 4

90°

0° (left) — 180° (right)

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

270°

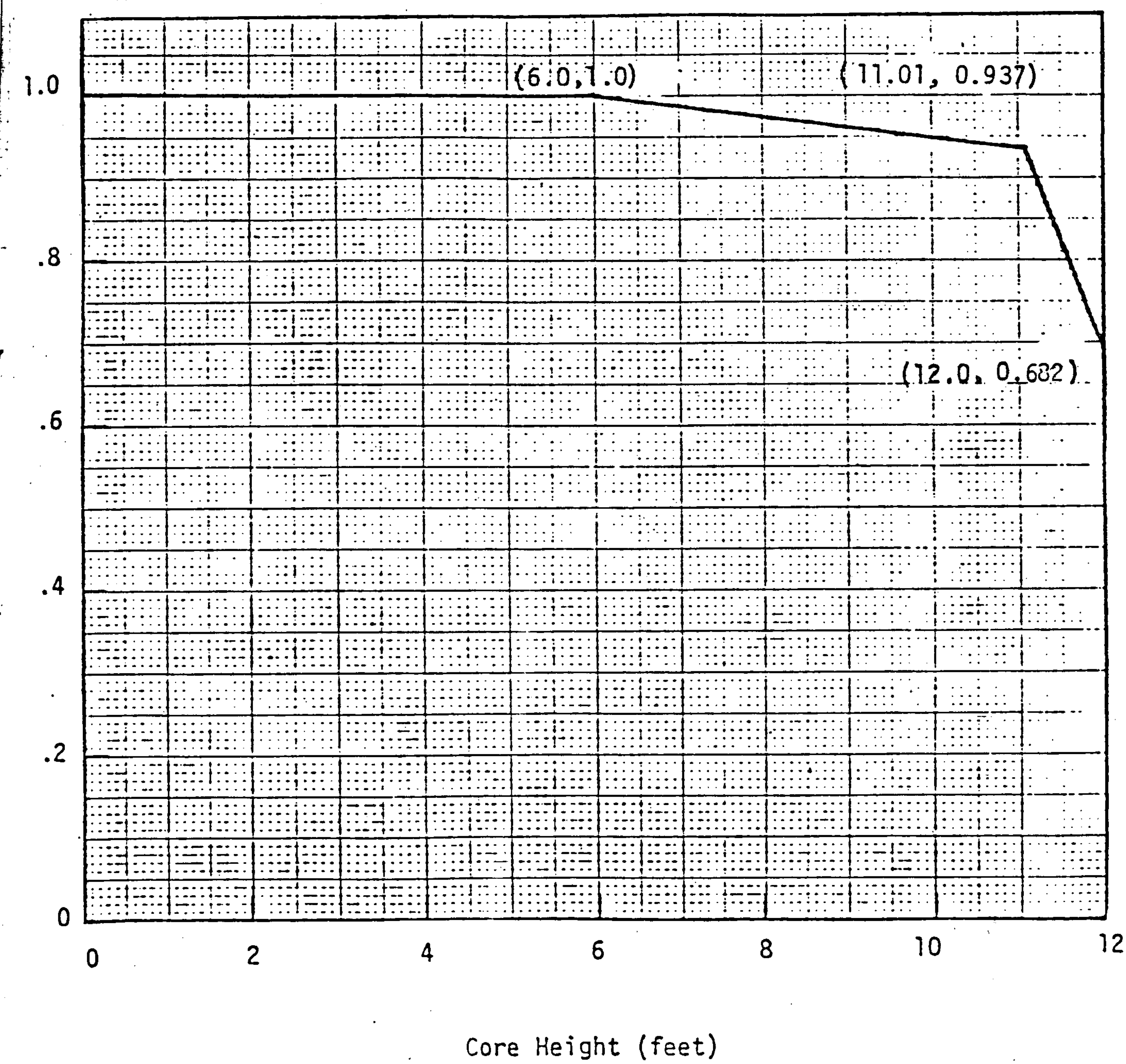
X	- Region Number
12	- Number of new burnable absorber rods
SS	- Secondary Source Rods

*	- Locations for possible insertion of burnable absorber demonstration rodlets (see Section 2.2)
12D	- 12 Depleted burnable absorber rods from Cycle 3

FIGURE 2

HEAT FLUX HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE
FOUR-LOOP OPERATION

Basis: $F_Q(Z) \times P$ ECCS limit of 2.20



123 Main Street
White Plains, New York 10601
914 681.6240

NewYorkPower Authority

J. Phillip Bayne
Executive Vice President
Nuclear Generation

June 9, 1983
IPN-83-56

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
ECCS Reanalysis

Dear Sir:

Westinghouse Electric Corporation has performed a new Appendix K ECCS analysis based on the NRC approved 1981 evaluation model considering 24% steam generator tube plugging. The new analysis demonstrates that Indian Point 3 complies with the requirements of 10 CFR 50.46 provided that total core peaking factor FQT does not exceed 2.20. Proposed Technical Specification changes reflecting this restriction have been submitted via our letter of May 5, 1983 (IPN-83-37). Forty copies of this analysis, entitled "Reload Safety Evaluation Indian Point Nuclear Power Plant Unit 3, Cycle 4 Revision 1", are enclosed for your review.

Enclosed with our letter of January 13, 1982 was a check in the amount of $4,000 for the review of the 24% steam generator tube plugging ECCS analysis and the previously mentioned proposed Technical Specification changes. Pursuant to 10 CFR 170.22, this issue has been determined to be a Class III item as it involves a single safety issue.

The 24% steam generator tube plugging ECCS analysis has been reviewed by the Authority's Plant Operating Review Committee and Safety Review Committee.

A001
1/1

Should you or your staff have any questions regarding this matter, please contact Mr. P. Kokolakis of my staff.

Very truly yours,

J. P. Bayne
Executive Vice President
Nuclear Generation

cc: Resident Inspector's Office
Indian Point Unit 3
U. S. Nuclear Regulatory Commission
P. O. Box 66
Buchanan, New York 10511