ATTACHMENT II

SAFETY EVALUATION

Power Authority of the
State of New York
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# RELOAD SAFETY EVALUATION
# INDIAN POINT NUCLEAR PLANT
# UNIT 3, CYCLE 2

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## 1.0 INTRODUCTION AND SUMMARY

The Indian Point Nuclear Power Plant, Unit Number 3, is in its first cycle of operation. The unit is expected to be refueled and ready for Cycle 2 startup in July, 1978.

This report presents an evaluation for Cycle 2 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 the fuel reload have been reviewed for the Cycle 2 design described herein. The results of new analyses have been included* and the justification for the applicability of previous results 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. This conclusion is based on the assumption that: (1) Cycle 1 operation is terminated after 17,100 ± 500 MWD/MTU, and (2) there is adherence to plant operating limitations discussed later in proposed modifications to the technical specifications.

During the Cycle 1/2 refueling, sixty four Region 1 and 1A fuel assemblies will be replaced by sixty four Region 4 fuel assemblies. Table 1 presents the number of assemblies in each region. The expected core loading pattern for Cycle 2 is shown in Figure 1.

Nominal design parameters for Cycle 2 are 3025 Mwth (100% rated core power), 2250 psia system pressure, 542.6°F vessel inlet temperature, and 6.24 kw/ft average linear power density (based on 144.0 inch active fuel length).

*New small break LOCA Analysis provided in Reference 9.

# 2.0 REACTOR DESIGN

## 2.1 MECHANICAL DESIGN

The mechanical design of Region 4 fuel is the same as Region 3 fuel except as noted below. The region enrichments are shown in Table 1. The Region 4 initial pre-pressurization level of the fuel rods is 35 psi less than Region 3. A minor change in fuel rod end plug design was made to increase the rod internal void volume. The Region 4 fuel has been designed according to the fuel performance model given in Reference 2. The fuel rod internal pressure will not exceed system pressure during Cycle 2.

Clad flattening time is predicted to be greater than 30,000 EFPH* for the limiting region (Region 2) using the current Westinghouse evaluation model[3]. Therefore, Region 2 has a Cycle 2 allowed residence time in excess of 18,000 EFPH. This is based upon a residence time for Cycle 1 of approximately 11,900 EFPH. Expected Cycle 2 lifetime is 7,800 EFPH.

Westinghouse has had considerable experience with Zircaloy-clad fuel. This experience is described in WCAP-8183, "Operational Experience with Westinghouse Cores,"[4] which is updated periodically.

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*EFPH = effective full power hours, integrated reactor power equivalent to operating at 100% for a stated time interval.

## 2.2 NUCLEAR DESIGN

The Cycle 2 core design is based on the Cycle 1 flux difference ($\Delta I$) band width ($\pm 5\%$) and the proposed Technical Specification changes provided in Section 4. Thus the Cycle 2 core loading is set to meet a $F_Q^T \times P$ ECCS limit of $\leq 2.32 \times K(Z)$ with K(Z) given in Figure 2 when utilizing the positive limit on axial flux difference and control rod insertion limits provided in Section 4.

Table 2 provides a comparison of the Cycle 2 kinetics characteristics with the current limit based on previously submitted accident analysis. It can be seen from the table that most of the Cycle 2 values fall within the current limits (the effect of the ones which don't is addressed in Section 3). Table 3 provides the control rod worths and requirements. The available shutdown margin exceeds the minimum required. Figure 3 provides control rod insertion limits which preserve shutdown margin and assure peaking factors are not exceeded during anticipated power control maneuvers.

Four Region 2 fuel assemblies contain burnable poison rods. Two of the fuel assemblies contain secondary source rod assemblies with their associated burnable poison rods. The other two fuel assemblies contain matching depleted burnable poison rods to maintain symmetry (see Figure 1).

The trip reactivity insertion rate for Cycle 2 is slower than the one used in previous analyses (see Section 3). The reactivity insertion rate is different because the combined bank worth as a function of time (axial location) has changed. The reactivity insertion rate for Cycle 2 was calculated by a very conservative method that produces a flux distribution skewed towards the bottom of the core. This reduces the reactivity worth of the banks at the top of the core relative to the total worth. Such a calculation provides a conservative trip reactivity shape.

## 2.3 THERMAL AND HYDRAULIC DESIGN

No significant variations in thermal margins will result from the Cycle 2 reload. The present DNB core limits which are given in the technical specifications 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 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. For the overpower transient, the fuel centerline temperature limit of 4700°F can be accommodated with margin during Cycle 2. The time dependent densification model was used for fuel temperature evaluations. The LOCA limit is met by maintaining $F_Q$ x P at or below 2.32 x K(Z) with 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] and, fuel densification report[5] 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 and fuel densification report are still valid.

A core reload can typically affect accident analysis input parameters in the following areas: core kinetic characteristics, control rod worths, and core peaking factors. Cycle 2 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 2 core physics parameters with current limits is given in Table 2. Except for delayed neutron fractions all the kinetics values remain within the bounds of the current limits. The minimum delayed neutron fraction, $\beta$, for the beginning and end of life Cycle 2 is outside the current limits. This will significantly affect only the control rod ejection transients, which are discussed in Section 3.3.

### 3.2.2 CONTROL ROD WORTHS

Changes in control rod worths may affect differential rod worths, shutdown margin, ejected rod worths, and trip reactivity. Table 2 shows that the maximum differential rod worth of two RCCA control banks moving together in their highest worth region for Cycle 2 is less than or equal to the current limit. Table 3 shows that the Cycle 2 shutdown margin requirements are satisfied. Ejected rod worths for Cycle 2 are within the bounds of the current limits.

Cycle 2 has a slower trip reactivity insertion rate than used in previous analyses. The effects of this reduced reactivity trip rate have been evaluated for those accidents affected, and compared with previous analyses. Slow transients are relatively insensitive to changes in trip reactivity insertion rate and, therefore, need not be re-analyzed 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, since the transient is essentially turned around before rod insertion starts. The effect of variations in trip reactivity insertion rates for the rod withdrawal at power incident has been investigated. The results of this analysis show that the minimum DNBR is unaffected, since the minimum DNBR for the transient occurs at relatively low reactivity insertion rates.

For the loss of flow and locked rotor transients the change in trip reactivity versus rod insertion will result in a slightly higher transient heat flux. Since the minimum DNB ratio for the loss of flow transient, and the minimum DNB ratio, peak clad temperature, and peak primary system pressure for the locked rotor transient, are sensitive to heat flux relative to flow, these incidents were reanalyzed. The results of these calculations are discussed in Section 3.3.

### 3.2.3 CORE PEAKING FACTORS

Evaluation of peaking factors for the rod out of position, dropped RCCA, and dropped bank incidents shows that DNBR is maintained above 1.3. Table 4 shows the peaking factors following control rod ejection. The peaking factors for the beginning and end-of-life, zero power rod ejection cases exceed the previous analysis values. These cases have been reanalyzed using the higher peaking factors, as discussed in Section 3.3.

The steamline break transients (FSAR Section 14.2.5) were reanalyzed for Cycle 2 using more recent analysis methods. The analyses and results are discussed in Section 3.3.

## 3.3 INCIDENTS REANALYZED

### Rod Ejection

For both the beginning and end of life rod ejection cases, the delayed neutron fractions are less than the current limits. Thus, all the rod ejection cases were reanalyzed to ensure that the fuel and clad limits were not exceeded. In addition, the beginning and end-of-life zero power cases were reanalyzed with increased peaking factors as shown in Table 4. The analysis was performed using the same methods as described

in References 5 and 6. The results are given in Table 5. The results show that the fuel rod at the hot spot does not exceed the limiting criteria. All criteria specified in Reference 6 continue to be met and, thus, the conclusions presented in Reference 5 are still valid.

Loss of Flow and Locked Rotor

The complete loss of flow and locked rotor transients were reanalyzed due to a slower trip reactivity insertion rate. The analyses were performed using the same methods and assumptions used for Cycle 1[5].

For the complete 4/4 pump loss of flow incident, the minimum DNB ratio does not fall below the limiting value of 1.30. The other cases described in the FSAR incident show larger DNB ratios than the 4/4 pump incident. Thus it is concluded that all cases would remain above the 1.30 limit and the conclusions as presented in the FSAR for this incident are still valid.

For the four and three loop operation locked rotor cases, Reference (5) showed that DNBR remained above 1.3. However, DNB was conservatively assumed to occur at the beginning of the locked rotor transient for the purpose of calculating the clad temperature transient. The four loop operation locked rotor case is most limiting from a peak clad temperature standpoint. The results show a peak clad temperature of 1801°F, well below the limiting value of 2700°F. The amount of $Zr/H_2O$ reaction is less than 1% by weight. The three loop operation locked rotor case is most limiting with respect to peak system pressure. The results show a peak system pressure of less than 2620 psia, which is below the value which would cause stresses to exceed the faulted condition stress limits of the primary coolant system. Thus, the conclusions as presented in the densification report and the FSAR are still valid for Cycle 2.

Steamline Break

The steamline break transients were reanalyzed for cycle 2. The reactor coolant system transient was calculated using the MARVEL Code (7). Initial conditions of power, temperature, pressure, flow and shutdown margin were consistent with the analysis presented in Chapter 14 of the FSAR(1). The steamline break accident assuming loss of offsite power was reanalyzed for cycle 2 using codes and methods used on previous submittals (eg. Trojan, Salem Unit 1), except for the method of calculating the Doppler Power coefficient. The cycle 2 coefficient properly accounts for the reduced reactor coolant flow which exists for the case with loss of offsite power, including the effects of local density variations as a function of flow rate and power level. Appropriate conservatism has been retained in the coefficient and transient. Other analysis assumptions were similar to those described in Reference (8). The results of the analysis show that for the hypothetical steamline break cases (complete severence of a main steam pipe), the minimum DNBR is greater than 1.3. For the credible break (a break equivalent to the spurious opening, with failure to close, of the largest of any single steam bypass, relief, or safety valve), the core does not return critical. Thus, all safety criteria are met and the conclusions presented in the FSAR are still valid.

## 3.4 ECCS ANALYSIS

The small break LOCA was reanalyzed for cycle 2 to accommodate an increase in the K(Z) third line coordinate (Figure 2). This analysis is presented in Reference 9. The requirements of 10CFR50.46 are met.

## 4.0 TECHNICAL SPECIFICATIONS

This section contains the technical content of proposed changes to the Indian Point Unit 3 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 3.10.2 - POWER DISTRIBUTION LIMITS

Replace Figure 3.10-2.

The increase in the K(Z) third line coordinate in Figure 2 from (12.0, 0.431) to (12.0, 0.647) 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.

Add to Section 3.10.2.4, "The indicated axial flux difference will be maintained less than + 12.5% at 100% power with the allowed axial flux difference increasing by 0.65% for each 1% reduction in power." This limit was used when verifying core peaking factor limits are met.

### 4.2 SPECIFICATION 3.10.4 - ROD INSERTION LIMITS

Revision: Replace Figure 3.10-4 with the attached Figure 3. This assures that core peaking factor limits are not exceeded during power control maneuvers allowed by the Technical Specifications.

## 5.0 REFERENCES


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

2. Miller, J. V. (Ed), "Improved Analytical Model Used In Westinghouse Fuel Rod Design Computations", WCAP-8785, October 1976.

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

4. Schreiber, R. E. and Iorii, J. A., "Operational Experience with Westinghouse Cores", WCAP-8183 Revision 6, June, 1977.

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

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

7. "MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System", WCAP-8844, November 1977.

8. "Reference Core Report 17 x 17", WCAP-8185, December, 1973.

9. Indian Point Unit 3 Small Break ECCS Analysis, February 1978.

TABLE 1

FUEL ASSEMBLY DESIGN PARAMETERS
INDIAN POINT UNIT NUMBER 3 - CYCLE 2

| Region | 1 | 2 | 3 | 4 |
|---|---|---|---|---|
| Enrichment (w/o of U 235)* | 2.28 | 2.80 | 3.29 | 3.10 |
| Density (percent theoretical)* | 94.6 | 94.5 | 94.4 | 94.5 |
| Number of Assemblies | 1 | 64 | 64 | 64 |
| Approximate Burnup at Beginning of Cycle 2 (MWD/MTU) | 16800** | 19000 | 13400 | 0 |

*Regions 1, 2, and 3 are as-built values. Region 4 values are those used for design analysis.

**Burnup of single Region 1 assembly used in Cycle 2, not entire Region 1 burnup at the end of Cycle 1.

TABLE 2

KINETICS CHARACTERISTICS
INDIAN POINT UNIT NUMBER 3 - CYCLE 2

| | Previous Analysis Values (1)(5) | Cycle 2 |
|---|---|---|
| Moderator Temperature Coefficient, (PCM/°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 $\beta_{eff}$ (percent) | .50 to .70 | 0.44 to .70 |
| Maximum Prompt Neutron Lifetime ($\mu$ sec) | 19 | $\leq$19 |
| Maximum Reactivity Insertion Rate for Two Banks Moving Together at HZP (PCM/SEC)* | 80 | $\leq$80 |

* PCM $\equiv 10^{-5}\ \Delta\rho$

**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 1 AND 2

| | Cycle 1 | | Cycle 2 | |
|---|---|---|---|---|
| | BOC | EOC | BOC | EOC |
| Control Rod Worth (percent $\Delta\rho$) | | | | |
| All Rods Inserted Less Worst Stuck Rod | 7.75 | 7.57 | 5.37 | 5.88 |
| (A) Less 10% | 6.97 | 6.81 | 4.83 | 5.29 |
| Control Rod Requirements (percent $\Delta\rho$) | | | | |
| Reactivity Defects (Doppler, Tavg, Void, Redistribution) | 1.81 | 2.93 | 1.64 | 2.67 |
| Rod Insertion Allowance | .50 | .50 | .50 | .50 |
| (B) Total Requirements | 2.31 | 3.43 | 2.14 | 3.17 |
| Shutdown Margin [(A)-(B)] (percent $\Delta\rho$) | 4.66 | 3.38 | 2.69 | 2.12 |
| Required Shutdown Margin (percent $\Delta\rho$)(1) | 1.0 | 1.72 | 1.0 | 1.72 |

TABLE 4

ROD EJECTION PARAMETERS
INDIAN POINT UNIT 3 CYCLE 2

| | Previous Analysis Values(5) | Cycle 2 | Values Used In Reanalysis |
|---|---|---|---|
| HZP-BOL | | | |
| Max. Ejected Rod Worth, %Δρ | 0.75 | ≤0.75 | 0.75 |
| Max. $F_Q^N$ | 12.7 | 14.2 | 14.2 |
| $\beta_{eff}$ | 0.007 | 0.0056 | 0.0055 |
| HFP-BOL | | | |
| Max. Ejected Rod Worth, %Δρ | 0.19 | ≤0.19 | 0.19 |
| Max. $F_Q^N$ | 5.22 | ≤5.22 | 5.22 |
| $\beta_{eff}$ | 0.007 | 0.0056 | 0.0055 |
| HZP-EOL | | | |
| Max. Ejected Rod Worth, %Δρ | 0.88 | <0.75 | 0.75 |
| Max. $F_Q^N$ | 15.1 | 17.1 | 17.1 |
| $\beta_{eff}$ | 0.005 | 0.0044 | 0.0044 |
| HFP-EOL | | | |
| Max. Ejected Rod Worth, %Δρ | 0.38 | ≤0.38 | 0.38 |
| Max. $F_Q^N$ | 5.78 | ≤5.78 | 5.78 |
| $\beta_{eff}$ | 0.005 | 0.0044 | 0.0044 |

TABLE 5

RESULTS OF ROD EJECTION ANALYSIS HOT SPOT FUEL AND CLAD TEMPERATURES

INDIAN POINT UNIT 3 CYCLE 2

| | BØL | BØL | EØL | EØL |
|---|---|---|---|---|
| Initial Power, % | 0 | 102 | 0 | 102 |
| Maximum Fuel Pellet Center Temperature (°F) | 3004 | 4697 | 3702 | 4886 |
| Maximum Fuel Average Temperature (°F) | 2605 | 3554 | 3262 | 3946 |
| Maximum Clad Temperature (°F) | 1974 | 2051 | 2470 | 2283 |
| Maximum Fuel Enthalpy (cal./gm) | 105.1 | 151.3 | 136.8 | 171.5 |

FIGURE 1

CORE LOADING PATTERN
INDIAN POINT UNIT 3 - CYCLE 2

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

90°

X - REGION NUMBER
Y - NUMBER OF BURNABLE POISON RODS

SS - SECONDARY SOURCE RODS
SS* - SPARE SECONDARY SOURCE (OPTIONAL)