UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 26, 1994

Docket No. 50-247

Mr. Stephen B. Bram
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, New York 10511

Dear Mr. Bram:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M88038)

The Commission has issued the enclosed Amendment No. 173 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated September 29, 1993, as supplemented by letter dated April 1, 1994.

The amendment revises the Technical Specifications to remove the cycle-specific parameter limits and to reference a Core Operating Limits Report (COLR) containing these limits. These changes are in accordance with the guidance provided in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications." Because plant operation will continue to be limited in accordance with the values of cycle-specific limits that are established using NRC-approved methodologies and the calculational methodologies and acceptance criteria are specified in the TS, the change is in conformance with 10 CFR 50.36.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Francis J. Williams, Jr., Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 173 to DPR-26
2. Safety Evaluation

cc w/enclosures:
See next page

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Consolidated Edison Company
of New York, Inc.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 173
License No. DPR-26

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Consolidated Edison Company of of New York, Inc. (the licensee) dated September 29, 1993, as supplemented by letter dated April 1, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 173, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance of the COLR by the licensee to be implemented no later than the return to operation following the 1995 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael Boyle

Michael L. Boyle, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 26, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 173

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
viii	viii
-	1-8
2.1-3	2.1-3
3.10-1	3.10-1
3.10-2	3.10-2
3.10-3	3.10-3
3.10-4	3.10-4
3.10-5	3.10-5
3.10-6	3.10-6
3.10-7	3.10-7
3.10-8	3.10-8
3.10-9	3.10-9
3.10-10	3.10-10
3.10-11	3.10-11
3.10-12	3.10-12
3.10-13	3.10-13
3.10-14	3.10-14
3.10-15	3.10-15
3.10-16	3.10-16
3.10-17	3.10-17
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-	6-29
-	6-30

LIST OF FIGURES

1.25 CORE OPERATING LIMITS REPORT (COLR)

The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specifications 6.9.1.8 - 6.9.1.11. Plant operation within these operating limits is addressed in individual specifications.

analyses using values of input parameters without uncertainties. In addition, margin is maintained by performing DNB design evaluations to a higher DNBR value, called the Safety Limit DNBR. This margin is sufficient to cover applicable rod bow DNB penalties and provide margin for use in design and operational flexibility.

The curves of Figure 2.1-1 show the loci of points of thermal power Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. These curves are based on a peak nuclear hot channel factor as stated in the Core Operating Limits Report (COLR) and a 1.55 cosine axial power shape.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the limits on core fission power distributions and to the limits on control rod operations.

Objectives

1. to ensure core subcriticality after reactor trip,

2. to ensure acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and

3. to limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications

3.10.1 Shutdown Reactivity

The shutdown margin shall be at least as great as that shown in Figure 3.10-1.

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low-power physics tests, the hot channel factors defined in the basis must meet the limits specified in the Core Operating Limits Report (COLR).

3.10.2.2 Following initial core loading, subsequent reloading and at regular effective full-power monthly intervals thereafter, power distribution maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of the COLR are satisfied. For the purpose of this comparison,

3.10.2.2.1 The measurement of total peaking factor, F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

3.10.2.2.2 The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by four percent to account for measurement error. If either measured hot channel factor exceeds its limit specified in the COLR, the reactor power and high neutron flux trip setpoint shall be reduced so as not to exceed a fraction of rated value equal to the ratio of the F_Q or $F^N_{\Delta H}$ limit to measured value, whichever is less. If subsequent in-core mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing.

3.10.2.3 The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per effective full-power quarter. The target flux difference must be updated each effective full-power month by linear interpolation using the most recent measured value and a value of approximately 0 percent at the end of the cycle life.

3.10.2.4 Except during physics tests, during excore calibration procedures and except as modified by Items 3.10.2.5 through 3.10.2.7 below, the indicated axial flux difference shall be maintained within the band specified in the COLR about the target flux difference (defines the band on axial flux difference).

3.10.2.5 At a power level greater than 90% of rated power, 3.10.2.5.1 If the indicated axial flux difference deviates from its target band, the flux difference shall be returned to its target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

3.10.2.6 At a power level no greater than 90 percent of rated power,

3.10.2.6.1 The indicated axial flux difference may deviate from its target band specified in the COLR for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference does not exceed an envelope bounded by that specified in the COLR at 90% power and increasing by the value specified in the COLR for each 2 percent of rated power below 90% power.

3.10.2.6.2 If Specification 3.10.2.6.1 is violated, then the reactor power shall be reduced immediately to no greater than 50% power and the high neutron flux setpoint reduced to no greater than 55 percent of rated values.

3.10.2.6.3 A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference being within its target band.

3.10.2.7 At a power level no greater than 50 percent of rated power,

3.10.2.7.1 The indicated axial flux difference may deviate from its target band.

3.10.2.7.2 A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24-hour period. One-half the time the indicated axial flux difference is out of its target band up to 50% of rated power is to be counted as contributing to the one hour cumulative maximum the flux difference may deviate from its target band at a power level ≤ 90% of rated power.

3.10.2.8 Alarms are provided to indicate non-conformance with the flux difference requirements of 3.10.2.5.1 and the flux difference-time requirements of 3.10.2.6.1. If the alarms are temporarily out of service, conformance with the applicable limit shall be demonstrated by logging the flux difference at hourly intervals for the first 24 hours and half-hourly thereafter.

3.10.2.9 If the core is operating above 75% power with one excore nuclear channel out of service, then core quadrant power balance shall be determined once a day using movable incore detectors (at least two thimbles per quadrant).

3.10.3 Quadrant Power Tilt Limits

3.10.3.1 Except for physics tests, when the core is operating above 50% of rated thermal power and the indicated quadrant power tilt ratio exceeds 1.02 but is less than or equal to 1.09, within two hours reduce the quadrant power tilt ratio to within its limit or the following actions shall be taken:

a. Restrict core power level and reset the power range high flux setpoint three percent of rated values for every percent of indicated power tilt ratio exceeding 1.0, and

b. Verify that the quadrant power tilt ratio is within its limit within 24 hours after exceeding the limit or restrict core power level to less than 50% of rated thermal power within the next 2 hours and reduce the power range high flux trip setpoint to less than or equal to 55% of rated thermal power within the next 4 hours.

3.10.3.2 Except for physics tests, if the indicated quadrant power tilt ratio exceeds 1.09 with the core operating above 50% of rated thermal power and

a) there is a simultaneous indication of a misaligned control rod, restrict core power level three percent of rated value for every percent of indicated power tilt ratio exceeding 1.0 or until core power level is less than 50% of rated thermal power. If the quadrant power tilt ratio is not within its limit within 2 hours after exceeding the limit, restrict core power level to less than 50% of rated thermal power within the next 2 hours and reduce the power range high flux trip setpoint to less than or equal to 55% of rated thermal power within the next 4 hours.

-or-

b) there is no simultaneous indication of a misaligned control rod, reduce thermal power to less than 50% of rated thermal power within 2 hours and reduce the power range high flux trip setpoint to less than or equal to 55% of rated thermal power within the next 4 hours.

3.10.3.3 The rod position indicators shall be monitored and logged once each shift to verify rod position within each bank assignment.

3.10.3.4 The tilt deviation alarm shall be set to annunciate whenever the excore tilt ratio exceeds 1.02, except as modified in Specification 3.10.10.

3.10.4 <u>Rod Insertion Limits</u>

3.10.4.1 The shutdown rods shall be withdrawn as specified in the COLR when the reactor is critical or approaching criticality (i.e., the reactor is no longer subcritical by an amount equal to or greater than the shutdown margin in Figure 3.10-1).

3.10.4.2 When the reactor is critical, the control banks shall be limited in physical insertion to the insertion limits specified in the COLR.

3.10.4.3 Control bank insertion shall be further restricted if:

a. The measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown,

b. A rod is inoperable (Specification 3.10.7).

3.10.4.4 Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-1 must be maintained except for the low-power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one control rod inserted.

3.10.5 <u>Rod Misalignment Limitations</u>

3.10.5.1.1 If a control rod is misaligned from its bank demand position by more than ±12 steps when indicated control rod position is less than or equal to 210 steps withdrawn, then realign the rod or determine the core peaking factors within 2 hours and apply Specification 3.10.2.

3.10.5.1.2 If a control rod is misaligned from its bank demand position by more than +17, -12 steps when indicated control rod position is greater than or equal to 211 steps withdrawn, then realign the rod or determine the core peaking factors within 2 hours and apply Specification 3.10.2.

3.10.5.2 If the restrictions of Specification 3.10.3 are determined not to apply and the core peaking factors have not been determined within two hours and the rod remains misaligned, the high reactor flux setpoint shall be reduced to 85% of its rated value.

3.10.5.3 If the misaligned control rod is not realigned within 8 hours, the rod shall be declared inoperable.

3.10.6 Inoperable Rod Position Indicator Channels

3.10.6.1 A rod position indicator channel shall be capable of determining control rod position within ±12 steps for indicated control rod position less than or equal to 210 steps withdrawn and +17, -12 steps for indicated control rod position greater than or equal to 211 steps withdrawn, or

a. For operation between 50 percent and 100 percent of rating, the position of the control rod shall be checked indirectly by core instrumentation (excore detectors and/or movable incore detectors) every shift, or subsequent to rod motion exceeding 24 steps, whichever occurs first.

b. During operation below 50 percent of rating, no special monitoring is required.

3.10.6.2 Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time.

3.10.6.3 If a control rod having a rod position indicator channel out of service is found to be misaligned from Specification 3.10.6.1a, above, then Specification 3.10.5 will be applied.

3.10.7 Inoperable Rod Limitations

3.10.7.1 An inoperable rod is a rod which does not trip or which is declared inoperable under Specification 3.10.5, or which fails to meet the requirements of Specification 3.10.8.

3.10.7.2 Not more than one inoperable control rod shall be allowed any time the reactor is critical except during physics tests requiring intentional rod misalignment. Otherwise, the plant shall be brought to the hot shutdown condition.

3.10.7.3 If any rod has been declared inoperable, then the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days. The analysis shall include due allowance for non-uniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

3.10.8 Rod Drop Time

At operating temperature and full flow, the drop time of each control rod shall be no greater than 2.4 seconds from gripper release to dashpot entry.

3.10.9 Rod Position Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift and after a load change greater than 10 percent of rated power.

3.10.10 Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs shall be logged once per shift and after a load change greater than 10 percent of rated power.

Basis

Design criteria have been chosen for normal operations, for operational transients and for those events analyzed in UFSAR Section 14.1 which are consistent with the fuel integrity analyses. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also the minimum DNBR in the core must be greater than the safety limits DNBRs in normal operation or in short-term transients.

In addition to the above conditions, the peak linear power density must not exceed the limiting kw/ft values which result from the large break loss-of-coolant accident analysis based on the ECCS acceptance criteria limit of 2200°F. This is required to meet the initial conditions assumed for a loss-of-coolant accident. To aid in specifying the limits on power distribution the following hot channel-factors are defined.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F^E_Q, Engineering Heat Flux Hot Channel Factor is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

The upper bound envelope of the total peaking factor (F_Q) specified in the COLR times the normalized peaking factor axial dependence of K(Z) specified in the COLR has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss-of-coolant accident analyses based on the specified F_Q times K(Z) specified in the COLR indicate a peak clad temperature of less than 2200°F for the worst case double-ended cold leg guillotine break[1].

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_{\Delta H}$ there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H}$ within the limits specified in the COLR. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods and can limit it to the desired value (he has no direct control over $F^N_{\Delta H}$) and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests at least each effective full-power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases, including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomalies which would otherwise affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 12 steps precludes rod misalignment no greater than 15 inches with consideration of maximum instrumentation error for indicated rod position less than or equal to 210 steps withdrawn.

For indicated control rod positions greater than or equal to 211 steps withdrawn, an indicated misalignment of +17 steps does not exceed the power peaking factor limits. The reactivity worth of a rod at this core height (211 + steps) is not sufficient to perturb power shapes to the extent that peaking factors are affected.

2. Control rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.4.

3. The control rod bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that, provided the above conditions (1 through 4) are observed, these hot channel factors limits are met. In the COLR, F_Q is arbitrarily limited for $P \leq 0.5$ (except for low-power physics tests).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full-power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup, but, expressed as axial offset, it varies only with burnup.

The technical specifications on power distribution control assure that the total peaking factor upper-bound envelope of specified F_Q times K(Z) as specified in the COLR is not exceeded and xenon distributions are not developed which, at a later time, would cause greater local power peaking even though flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the control rod bank more than 190 steps withdrawn (i.e., normal full-power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full-power at which the core was operating, is the full-power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full-power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated axial flux difference deviation as specified in the COLR is permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during part-power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations; this is acceptable due to the low probability of a significant accident occurring during these durations.

In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target bank when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target bank; however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range specified in the COLR. Therefore, while the deviation exists, the power level is limited to 90 percent or less depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the limit specified in the COLR for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full-power condition as possible. This is accomplished by using the boron system to position the control rods to produce the required indicated flux difference.

For Condition II events, the core is protected from overpower and a minimum DNBR of less than the safety limit DNBRs by an automatic protection system. Compliance with operating procedures is assumed as a precondition for Condition II transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Quadrant power tilt limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation as this phenomenon is caused by some asymmetric perturbation, e.g., rod misalignment or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication system or core instrumentation per Specification 3.10.6, and core limits are protected per Specification 3.10.5. A quadrant tilt by some other means would not appear instantaneously but would build up over several hours, and the quadrant tilt limits are met to protect against this situation. They also serve as a backup protection against the dropped or misaligned rod. Operational experience shows that normal power tilts are less than 1.01. Thus, sufficient time is available to recognize the presence of a tilt and correct the cause before a severe tilt could build up. During startup and power escalation, however, a large tilt could be indicated. Therefore, the specification has been written so as to prevent escalation above 50 percent power if a large tilt is present. The numerical limits are set to be commensurate with design and safety limits for DNB protection and linear heat generation rate as described below.

The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions are measured as part of the startup physics testing and are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses. It is not intended that reactor operation would continue with a power tilt condition which exceeds the radial power asymmetry considered in the power capability analysis.

The quadrant tilt power deviation alarm is used to indicate a sudden or unexpected change from the radial power distribution mentioned above. The two percent tilt alarm setpoint represents a minimum practical value consistent with instrumentation errors and operating procedures. This asymmetry level is sufficient to detect significant misalignment of control rods. Misalignment of control rods is considered to be the most likely cause of radial power asymmetry. The requirement for verifying rod position once each shift is imposed to preclude rod misalignment which would cause a tilt condition of less than the 2% alarm level.

The two-hour time interval in this specification is considered ample to identify a dropped or misaligned rod and complete realignment procedures to eliminate the tilt condition. In the event that this tilt condition cannot be eliminated within the two-hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full-core physics map utilizing the movable detector system. For a tilt condition ≤ 1.09, an additional 22-hour time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of three percent for each one percent of indicated tilt is required. Physics measurements have indicated that the core radial power peaking would not exceed a two to one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment.

In the event a tilt condition of ≤ 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to less than 50% of rated power. To avoid reset of a large number of protection setpoints, the power range nuclear instrumentation would be reset to cause an automatic reactor trip at 55% of allowed power. A reactor trip at this power has been selected to prevent, with margin, exceeding core safety limits even with a nine percent tilt condition.

If a tilt ratio greater than 1.09 occurs, which is not due to a misaligned rod, the reactor power level will be reduced to less than 50% of rated power for investigation. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (3% for each 1 percent the tilt ratio exceeds 1.0) for two hours to correct the rod misalignment.

Trip shutdown reactivity is provided consistent with plant safety analysis assumptions. One percent shutdown is adequate except for steam break analysis, which requires more shutdown if the boron concentration is low. Figure 3.10-1 is drawn accordingly.

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequence of a hypothetical rod ejection accident. The available control rod reactivity, or excess beyond needs, decreases with decreasing boron concentration because the negative reactivity required to reduce the power level from full power to zero power is largest when the boron concentration is low.

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.4) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod, that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest such as end-of-life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low-power and zero-power testing is permitted because of the brief period of the test and because special precautions are taken during these tests.

The rod position indicator channel is sufficiently accurate to detect a rod ±7.5 inches away from its demand position for indicated control rod position less than or equal to 210 steps withdrawn. An indicated misalignment ≤ 12 steps does not exceed the power peaking factor limits. A misaligned rod of + 17 steps allows for an instrumentation error of 12 steps plus 5 steps that are not indicated due to the location relationship of the RPI coil stack and the control rod drive rod for indicated rod position greater than or equal to 211 steps withdrawn. The last five steps of rod travel are not indicated by the RPI because the drive rod and spider assembly have been raised three inches (≈5 steps) from rod bottom. The reactivity worth of a rod at this core height (211 + steps) is not sufficient to perturb power shapes to the extent that peaking factors are affected. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors and/or movable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 12-step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully-inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 4 week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

The required drop time to dashpot entry is consistent with safety analysis.

Reference

1. UFSAR Section 14.3

6.9.1.8 Core operating limits shall be established and documented prior to each reload cycle, or prior to any remaining portion of the cycle, for the following:

a. Axial Flux Difference limits for Specifications 3.10.2.

b. Height Dependent Heat Flux Hot Channel Factor for Specification 3.10.2.

c. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3.10.2.

d. Shutdown Bank Insertion Limit for Specification 3.10.4.

e. Control Bank Insertion Limits for Specification 3.10.4.

6.9.1.9 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specification 3.10.4 - Shutdown Bank Insertion Limit, Control Bank Insertion Limits and 3.10.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary). (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

c. T.M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

d. NUREG-0800, Standard Review Plan, US Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

e. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary). (Methodology for Specification 3.10.2 Height Dependent Heat Flux Hot Channel Factor.)

6.9.1.10 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

6.9.1.11 The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

Special Reports

6.9.2 Special reports shall be submitted to the NRC Regional Administrator of the Region I Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Each containment integrated leak rate test shall be the subject of a summary technical report including results of the local leak rate test since the last report. The report shall include analyses and interpretations of the results which demonstrate compliance in meeting the leak rate limits specified in the Technical Specifications.

b. Inoperable fire protection and detection equipment (Specification 3.13).

c. Sealed source leakage in excess of limits (Specification 4.15).

d. The complete results of the steam generator tube inservice inspection (Specification 4.13.C.).

e. Radioactive effluents (Specification 3.9).

f. Radiological environmental monitoring (Specification 4.11).

g. Meteorological monitoring instrumentation (Specification 3.15).

h. Inoperable radiation and hydrogen monitoring instrumentation (Specification 3.5) outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

i. Operation of overpressure protection system (Specification 3.1.A.4).

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time intervals at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. Reportable Event Reports.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all source material on record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Updated Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material releases to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

g. Records of training and qualification for current members of the plant staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual except as noted in 6.10.1.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the SNSC and the NFSC.

l. Records for Environmental Qualification which are covered under the provisions of Specification 6.13.

m. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

n. Records of the service lives of all snubbers addressed by Section 3.12 of the Technical Specifications, including the date at which the service life commences and associated installation and maintenance records.*

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

* The documentation referred to herein is required for all snubbers beginning with those replaced following the issuance of Amendment 112.

6.12 HIGH RADIATION AREA

6.12.1 As an acceptable alternative to the "control device" or "alarm signal" required by 10 CFR 20.203(c)(2):

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of Specification 6.12.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry to such areas and the keys shall be maintained under the administrative control of the Watch Supervisor on duty.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines), or NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of License No. DPR-26 dated October 24, 1980.

6.13.2 By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines of NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information,

   b. a determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes, and

   c. documentation of the fact that the change has been reviewed and found acceptable by the SNSC.

2. Shall become effective upon review and acceptance by the SNSC.

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.15.1 The ODCM shall be approved by the Commission prior to implementation.

6.15.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

    a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluation justifying the change(s),

    b. a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations, and

    c. documentation of the fact the change has been revised and found acceptable by the SNSC.

2. Shall become effective upon review and acceptance by the SNSC.

6.16 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE SYSTEMS

6.16.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid) shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change was made. The discussion of each change shall contain:

a. a summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59,

b. sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information,

c. a detailed description of the equipment, components and processes involved and the interfaces with other plant systems,

d. an evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto,

e. an evaluation of the change, which shows the expected maximum exposures to individuals in the Unrestricted Area and to the general population that differ from those previously estimated in the license application and amendments thereto,

f. a comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period in which the changes are to be made;

g. an estimate of the exposure to plant operating personnel as a result of the change, and

h. documentation of the fact that the change was reviewed and found acceptable by the SNSC.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 173 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated September 29, 1993, as supplemented by letter dated April 1, 1994, the Consolidated Edison Company of New York (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 Technical Specifications (TSs). The requested changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to the Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of the TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter (GL) 88-16, dated October 4, 1988. The April 1, 1994, submittal was administrative in nature and identified the applicable TS section for each analytical method. It did not change the initial proposed no significant hazards consideration and was not outside the scope of the original Federal Register notice.

2.0 EVALUATION

Section 50.36 of Title 10 of the Code of Federal Regulations (CFR) established the regulatory requirements related to the content of TS. The rule requires that TSs include items in specific categories, including safety limits, limiting conditions for operation, and surveillance requirements; however, the rule does not specify the particular requirements to be included in a plant's TSs. The NRC developed criteria, as described in the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," (58 FR 39132) to determine which of the design conditions and associated surveillances need to be located in the TSs.

The Final Policy Statement adopted the subjective statement of the Atomic Safety and Licensing Appeal Board, ALAB 531, 9 NRC 263 (1979), (Trojan Nuclear Plant) as the basis for the criteria. The Appeal Board stated,

"... there is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a Technical Specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations in the Technical Specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an event giving rise to an immediate threat to the public health and safety."

Briefly, the criteria provided by the Final Policy Statement are: (1) detection of abnormal degradation of the reactor coolant pressure boundary, (2) boundary conditions for design basis accidents and transients, (3) primary success paths to prevent or mitigate design basis accidents and transients, and (4) functions determined to be important to risk or operating experience. The Commission's Final Policy Statement acknowledged that its implementation may result in the relocation of existing technical specification requirements to licensee controlled documents and programs.

The licensee's proposed changes to the TS are in accordance with the guidance provided by GL 88-16 and are addressed below.

(1) The Definition section of the TS was modified to include a definition of the COLR that requires cycle-specific parameter limits to be established on a unit-specific basis in accordance with NRC-approved methodologies that maintain the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

    (a) Specification 3.10.2.1

    The limits for the hot channel factors as defined in the basis for this specification are specified in the COLR.

    (b) Specification 3.10.2.2

    The limits for the comparison of measured hot channel factors for this specification are specified in the COLR.

(c) Specification 3.10.2.2.2

The limits for the total peaking factor and the enthalpy rise hot channel factor for this specification are specified in the COLR.

(d) Specification 3.10.2.4 and 3.10.2.6.1

The axial flux difference limits and the target bands for these specifications are specified in the COLR.

(e) Specification 3.10.4.1

Withdrawal limitations on the shutdown rods for this specification are specified in the COLR.

(f) Specification 3.10.4.2

The control bank insertion limits for this specification are specified in the COLR.

The Bases of these specifications have been modified by the licensee to include appropriate reference to the COLR.

(3) Specification 6.9.1.8 has been added as additional reporting requirements of the Administrative Controls section of the TS. This specification requires that the COLR be documented before each reload cycle or any remaining part of a reload cycle if specified parameters change, and, upon issuance, submitted to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, these specifications require that the values of these limits be established using NRC-approved methodologies. The approved methodologies and the TS sections to which they apply are as follows:

(a) WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specification 3.10.4 - Shutdown Bank Insertion Limit, Control Bank Insertion Limits and 3.10.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

(b) WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES-TOPICAL REPORT," September 1974 (W Proprietary). (Methodology for Specification 3.10.2 - Axial flux Difference (Constant Axial Offset Control).)

(c) T.M. Anderson to K.Kneil (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis

Aspects of an Improved Load Follow Package. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

(d) NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

(e) WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL - 1981 VERSION," February 1982 (W Proprietary). (Methodology for Specification 3.10.2 - Height Dependent Heat Flux Hot Channel Factor.)

The specific values of the limits in the IP2 COLR will be modified through the 10 CFR 50.59 process when such values are developed using NRC-approved methodologies consistent with all applicable limits of the safety analyses addressed in the IP2 Final Safety Analysis Report. As required by added TS Section 6.9.1.8, the COLR will be provided to the Commission upon issuance before each reload cycle or if specified parameters change during a fuel cycle. Plant operation will continue to be limited in accordance with the values of cycle-specific limits that are established using NRC-approved methodologies and the calculational methodologies and acceptance criteria are specified in the TS.

The staff's review of the proposed changes determined that the relocation of cycle specific parameter limits is administrative in nature and there is no impact on plant safety as a consequence. Although these limits are relocated from the TS to the COLR, the licensee must continue to evaluate any changes in accordance with 10 CFR 50.59. Should the licensee's determination conclude that an unreviewed safety question is involved, due to either: (1) a significant increase in the probability or consequences of accidents or malfunctions of equipment important to safety; (2) the creation of a possibility for an accident or malfunction of a different type than any evaluated previously; or (3) a significant reduction in the margin of safety, NRC approval and a license amendment would be required prior to implementation of the change. NRC inspection and enforcement programs also enable the staff to monitor facility changes and licensee adherence to NRC regulations and licensee commitments and to take any remedial action that may be appropriate.

The staff's review concluded that 10 CFR 50.36 does not require cycle-specific parameter limits to be retained in the TS. However, the staff determined that the inclusion of the limits are an operational detail related to the licensee's safety analyses which are adequately controlled by the requirements of 10 CFR 50.59. Therefore, the continued processing of license amendments related to revisions of the affected limits, where the revisions to those requirements do not involve an unreviewed safety question under 10 CFR 50.59, would afford no significant benefit with regard to protecting the public health and safety.

The staff has concluded, therefore, that relocation of the cycle specific parameter limits is acceptable because: (1) their inclusion in technical specifications is not specifically required by 10 CFR 50.36 or other regulations; (2) the limits have been relocated to the COLR, are adequately controlled by 10 CFR 50.59, and their inclusion in the TS is not required to avert an immediate threat to the public health and safety; and (3) changes which involve an unreviewed safety question, will require prior NRC approval through an application for an amendment pursuant to 10 CFR 50.90.

The List of Figures, page viii, was also modified to reflect the replacement of cycle-specific parameters with references to the COLR. This change is purely administrative and is, therefore, found to be acceptable.

3.0 <u>STATE CONSULTATION</u>

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 <u>ENVIRONMENTAL CONSIDERATION</u>

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 62154). The amendment also changes reordkeeping or reporting requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 <u>CONCLUSION</u>

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor:
F. Williams

Date: July 26, 1994

July 26, 1994

Docket No. 50-247

Mr. Stephen B. Bram
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, New York 10511

Dear Mr. Bram:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M88038)

The Commission has issued the enclosed Amendment No. 173 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated September 29, 1993, as supplemented by letter dated April 1, 1994.

The amendment revises the Technical Specifications to remove the cycle-specific parameter limits and to reference a Core Operating Limits Report (COLR) containing these limits. These changes are in accordance with the guidance provided in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications." Because plant operation will continue to be limited in accordance with the values of cycle-specific limits that are established using NRC-approved methodologies and the calculational methodologies and acceptance criteria are specified in the TS, the change is in conformance with 10 CFR 50.36.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original signed by:
Francis J. Williams, Jr., Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 173 to DPR-26
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	PDI-1:LA	PDI-1:PM	OGC w/noted revision	SRXB	PDI-1:D
NAME	CVogan	FWilliams:avl	CMarco	~~RCJones~~ JCollins	MBoyle
DATE	6/22/94	6/22/94	7/11/94	6/23/94	7/20/94

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DATED: July 26, 1994

AMENDMENT NO. 173 TO FACILITY OPERATING LICENSE NO. DPR-26-INDIAN POINT UNIT 2

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
M. Boyle
C. Vogan
F. Williams
OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
C. Grimes, 11/F/23
F. Williams
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list