April 28, 1995

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

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M89650)

Dear Mr. Quinn:

The Commission has issued the enclosed Amendment No. 182 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 (TSs) in response to your application transmitted by letter dated June 1, 1994, as supplemented by letters dated January 25, 1995, April 7, April 19, and April 26, 1995.

The amendment revises the TS Section 3.10 to allow extended Rod Position Indication (RPI) deviation limits and on-line calibration of the RPI channels for cycle 13 only.

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

Sincerely,

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

Docket No. 50-247

Enclosures: 1. Amendment No. 182 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\IP2\IP289650.AMD
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PDI-1:LA	E	PDI-1:PM	E	PDI-1:PM	E	OGC	N	PDI-1:D	
NAME	SLittle		DMcDonald:dt		FWilliams				LMarsh	
DATE	4/26/95		04/26/95		04/26/95		4/28/95		4/28/95	

OFFICIAL RECORD COPY

DF01 cP-1

9505040287 950428
PDR ADOCK 05000247
P PDR

DATED: April 28, 1995

AMENDMENT NO. 182 TO FACILITY OPERATING LICENSE NO. DPR-26-INDIAN POINT UNIT 2

DISTRIBUTION:
Docket File
PUBLIC
PDI-1 Reading
S. Varga, 14/E/4
J. Zwolinski, 14/H/3
L. Marsh
S. Little
F. Williams
OGC
D. Hagan, T-4 A43
G. Hill (2), T-5 C3
C. Grimes, 11/E/22
M. Chatterton 8E23
ACRS (4)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

000033

April 28, 1995

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

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M89650)

Dear Mr. Quinn:

The Commission has issued the enclosed Amendment No. 182 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 (TSs) in response to your application transmitted by letter dated June 1, 1994, as supplemented by letters dated January 25, 1995, April 7, April 19, and April 26, 1995.

The amendment revises the TS Section 3.10 to allow extended Rod Position Indication (RPI) deviation limits and on-line calibration of the RPI channels for cycle 13 only.

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

Docket No. 50-247

Enclosures: 1. Amendment No. 182 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\IP2\IP289650.AMD

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PDI-1:LA E	PDI-1:PM E	PDI-1:PM E	OGC N	PDI-1:D
NAME	SLittle	DMcDonald:dt	FWilliams	[illegible]	LMarsh
DATE	4/26/95	04/26/95	04/26/95	4/28/95	4/28/95

OFFICIAL RECORD COPY

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

April 28, 1995

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

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M89650)

Dear Mr. Quinn:

The Commission has issued the enclosed Amendment No. 182 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 (TSs) in response to your application transmitted by letter dated June 1, 1994, as supplemented by letters dated January 25, 1995, April 7, April 19, and April 26, 1995.

The amendment revises the TS Section 3.10 to allow extended Rod Position Indication (RPI) deviation limits and on-line calibration of the RPI channels for cycle 13 only.

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,

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

Docket No. 50-247

Enclosures: 1. Amendment No. 182 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

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. 182
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 June 1, 1994, as supplemented by letters dated January 25, 1995, April 7, April 19, and April 26, 1995, 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. 182, 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 its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

G E Edison for

Ledyard B. Marsh, 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: April 28, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 182

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
ii	ii
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
3.10-18	3.10-18
3.10-19	3.10-19
3.10-20	3.10-20

TABLE OF CONTENTS (Cont'd)

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. *In addition, insertion limits do not apply when performing calibration of individual rod position indicator channels at or below the rating specified in the Core Operating Limits Report (COLR) but not higher than a nominal 30% power not to exceed 35% power. 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.

* Only for Cycle 13.

3.10.5 Rod Misalignment Limitations

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 as follows: for operation at or below the rating specified in the Core Operating Limits Report (COLR) but no higher than 50% power within ±24 steps*; for operation above the rating, 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. During calibration a rod position indication channel is not considered to be inoperable.

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.

* Only for Cycle 13.

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. At or below the rating specified in the Core Operating Limits Report (COLR) but no higher than 50% power the rod position indicator capability is less than or equal to 24 steps.

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

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

5. 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.

Insertion limits do not apply during calibration of RPIs at or below the rating specified in the Core Operating Limits Report (COLR) but no higher than a nominal 30% power not to exceed 35% power because performing these calibrations at this reduced power ensures that the power peaking factor limits are met.

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 primary means of determining the position of individual control rods is the Rod Position Indication system. The RPI system consists of an individual rod position detector mounted on the pressure housing of each of the rod drive mechanisms, rack mounted electronic equipment and indicating equipment mounted on the flight panel. The rod position detector is a linear variable transformer consisting of primary and secondary coils alternatively stacked on a stainless steel support tube. The mechanism drive shaft serves as a "core" of the transformer. With a constant AC source applied to the primary windings, the vertical position of the mechanism drive rod shaft changes the primary to secondary magnetic coupling and produces a unique AC secondary voltage. This output voltage is an analog signal which is proportional to the vertical position of the control rod. The AC output from the secondary coils is fed to the signal conditioning circuit on the rod position chassis where is it rectified to a DC signal and filtered. The resulting DC analog voltage which is proportional to rod position is fed to the following points.

a) Rod bottom bistable
b) Flight panel indicator
c) Position voltmeter on flight panel
d) Test points on front of chassis
e) Plant Computers

A zero and span adjustment is provided to produce an output voltage signal proportional to rod travel between rods full in and rods full out. Because there is only a zero and span adjustment, a two point calibration is done.

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.

Experience at Indian Point 2 and at other plants with similar RPI systems has shown that the output signal of the RPI is not exactly linear with respect to vertical position of the control rod. Thus, there is some inherent error initially in the RPI indication. However, by calibrating the shutdown bank and control banks A, B and C at the fully withdrawn position, and control bank D at its normal operating position, the calibration will be most accurate at the position where the rods are usually found. In addition, experience has shown that the proportionality constant is sensitive to temperatures.

As a result of the above an additional uncertainty is added to the normal measurement uncertainty. To account for these uncertainties, data points can be collected and an individual graph for each RPI can be provided to the operator. As an alternative to individual graphs, a larger total uncertainty can be assumed for the RPI along with an equivalent assumed misalignment of a rod from the bank demand position. Calculations have been done that demonstrate that a total of ±24 steps can be tolerated as an error at or below the reduced power level given in the COLR but no more than 50% power. Since at some power levels it is not possible to determine whether there is rod motion or the RPI has drifted or is inaccurate, the calculations have assumed in the worst case a misalignment of 48 steps between a D bank control rod and the remainder of its group (i.e., 24 steps due to the RPI indication and 24 steps misalignment). This was also done for the C Bank (both banks were nominally at their 100% power insertion limits). For conservatism the Technical Specifications on allowed rod misalignment has been kept at ±12 steps, that is, for power levels where the rod position can be determined more accurately. If the indicated misalignment of ±24 steps has been exceeded, and a check has shown that the control rod(s) are indeed misaligned by more than ±12 steps, then the rod would be returned to ±12 steps or additional action must be taken as prescribed in the Technical Specification.

It is recognized that during certain reactor conditions the actual rod position cannot be determined. For example, during startup (subcritical) when the shutdown banks are withdrawn there may be misalignment, but because the reactor is subcritical, no independent verification possible. Therefore, the operator must rely on the RPI's. But, on the other hand, because there is no power, rod misalignment is of no significance. Therefore, the ±24 steps criteria for the RPI indication, when applied to actual rod misalignment would have no affect on thermal margins because of higher peaking factors. No increase in power is allowed until all shutdown banks are out, control bank A is out and control Banks B, C, and D are at or above the insertion limit.

Another situation where the actual rod position cannot be determined is when the reactor is being shutdown. Again for the control rods to be inserted beyond the insertion limit requires that the reactor be brought subcritical and again, rod misalignment would have no effect on thermal margins.

If it is determined that the RPI is out of calibration, on-line calibration of the instrumentation can be performed at or below the reduced power level given in the COLR but no higher than a nominal 30% power not to exceed 35% power. Thermal margins are maintained by reducing power to or below the respective COLR values for extended RPI deviation limits and on-line calibration.

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 24-step misalignment would have no significant 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.

<u>Reference</u>

1. UFSAR Section 14.3

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 182 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 June 1, 1994, as supplemented by letters dated January 25, 1995, April 7, April 19, and April 26, 1995, the Consolidated Edison Company of New York (Con Edison or the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 Technical Specifications (TSs). The requested changes would revise TS Section 3.10 to allow extended Rod Position Indication (RPI) deviation limits and on-line calibration of the RPI channels. The January 25, 1995, April 7, April 19, and April 26, 1995, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination and was within the scope of the original Federal Register notice.

Specifically, the initial proposed changes would have allowed extended RPI deviation limits of ±24 steps for power levels up to 85% power and on-line calibration of the RPI channels for power levels not to exceed 50%. As detailed in the evaluation that follows, Con Edison limited its request to allow the RPI deviation limits of ±24 steps for power levels up to 50% not 85% as initially requested and on-line calibration of the RPI channels at a nominal power level of 30% (not to exceed 35%) not the 50% requested. The April 26, 1995 submittal limited the request for use in cycle 13 only. Thus, the approved changes are within the original FEDERAL REGISTER notice.

The RPI system at Indian Point Unit No. 2 (IP2) provides the actual position (axial elevation) of each rod cluster control assembly (RCCA) relative to the bank demand position. The present TSs for IP2 permits deviations of ±12 steps (±7.5 inches) between the RPI channel output and the bank demand position over most of the range from fully inserted to fully withdrawn. During plant startup, particularly from the cold condition, the RPI channels may be subject to instabilities and drift until the control rod drive assemblies come to thermal equilibrium at operating temperature. These thermal instabilities cause indications that the RCCAs are misaligned from the bank demand position when in fact there is no actual deviation between actual RCCA position and the bank demand position. If such deviations indicate that there is more than a ±12 step misalignment in more than one channel per RCCA group or two channels per RCCA bank, the TSs require that the reactor be brought subcritical and the deviating RPI channels recalibrated. This process involves fully inserting RCCAs followed by withdrawal of the RCCAs with deviating RPIs. During withdrawal, the RPI signal is measured and recorded as a function of RCCA position to produce a new calibration.

In order to eliminate the loss of availability of IP2, Con Edison proposes an extended RPI deviation band and a procedure to allow on-line calibration of deviating RPI channels. The report submitted in support of this proposal describes the analyses and evaluation which were conducted to demonstrate the application of an extended RPI deviation band for lower powers and the on-line calibration of the RPI channels at low power. The NODE-P2 code was used to calculate the impact on core peaking factors of the extended deviation band. The calculated core peaking factors were compared with limits in the IP2 TSs to justify an extended band for powers less than 85%. In the same manner the NODE-P2 model was used to calculate the core peaking factors during on-line calibration of the RPI channels. Calculated values were compared with the IP2 TS limits at lower power levels to justify on-line calibration. Calculated core peaking factors were found to be acceptable for a power level of 50%.

In order to assure that the on-line calibration would not affect fuel reliability, fuel rod thermal mechanical duty during on-line calibration was evaluated and assessed. For this evaluation Con Edison used the FRAPCON computer code to model those fuel rods which are subject to the greatest power cycling during the RCCA insertion and withdrawal. The analyses demonstrated that on-line calibration of the RPI channels would have insignificant impact on fuel rod thermal mechanical duty. Furthermore, all postulated plant accidents and transients described in Chapter 14 of the IP2 Final Safety Analysis Report (FSAR) were reviewed and evaluated. This review focused on the potential consequences in the event that the accident or transient were initiated during calibration. The results showed that the consequences were no more severe than those analyzed in Chapter 14 of the FSAR.

2.0 EVALUATION

2.1 Rod Position Indicators and Power Distribution Limits

The IP2 TSs allow deviations between the RPI signal and the bank demand position of ± 12 steps for RCCA positions < 211 and - 12 to + 17 for RCCA positions >210 steps withdrawn. (This allows for an error in the sensing electronics of + 12 steps plus allowance for 5 steps which are not indicated due to the relationship of the RPI coil stack and the RCCA drive rod for indicated rod positions > 210 steps withdrawn.) The bases for the allowable deviation are the analyses of core power distributions under both steady state and anticipated transient conditions which are performed as part of the reload safety analysis for each fuel cycle. These analyses demonstrate that core peaking factor limits will not be exceeded under all anticipated steady state operating conditions and normal operational transients as permitted by the operating mode specified in the TSs, provided that no RCCA is misaligned from its bank by more than ± 12 steps. Compliance with the core peaking factors limits assures that the consequences of all postulated accidents as evaluated in the FSAR will be acceptable.

Implicit in the calculations for a fuel cycle is the assumption that the control rods in a particular bank move simultaneously and that all rods within the bank are in alignment within ± 12 steps. Misalignments in excess of ± 12 steps have the potential to increase the peaking factors. Thus, any increase in the range of permissible deviation between RPI signals and the bank demand position would have to be assessed in terms of the peaking factor limits in the TSs. The local core peaking factors are not routinely monitored during normal power operation. Surveillance of core power distribution is via the excore detectors which provide the core axial flux difference and core quadrant tilt.

2.2 Assessment of RPI Deviation Limits

For this analysis, Con Edison adopted the approach that the indicated misalignments represented actual rod misalignments and then calculated the resulting impact on core peaking factors and global core power distributions. The intention was to demonstrate that, even if the misalignments were actual, the resulting impact on core peaking is small and can be accommodated by limiting core power levels. Cycles 11, 12, and 13 were evaluated with respect to an extended RPI deviation band of ± 24 steps (15 inches). The effect of RCCA misalignment on core peaking factors was analyzed at beginning of cycle(BOC), mid cycle (MOC) and end of cycle (EOC). A large number of combinations of misaligned RCCAs were simulated at each of the three burnups with misalignments of up to ± 24 steps. These misalignments included individual RCCAs, symmetric and asymmetric groups of RCCAs and banks. The results of the RCCA misalignment analyses were summarized in tables which contain the maximum fractional change in nodal F_q and pin F delta H for each of the classes of misaligned rods. The fractional change in both of these peaking factors in relation to the current TS limit of ± 12 steps is shown. Con Edison proposed extending the ± 24 step limit to 85% power; however, the NRC staff did not consider the benchmarking of the NODE-P2 code (this will be discussed in Section 2.4) adequate to justify this high a power level, so the 50% power level was considered. The TS limit for F_q at 50% power is double that at 100% power. Since the maximum calculated fractional change in F_q is far less than 1, the misalignments of up to ± 24 steps is acceptable for power levels less than 50%. Similarly the allowed F delta H for 50% power is much greater than that calculated in the analysis. Based on the results of the analysis, Con Edison concluded that power operation with a RPI deviation limit of ± 24 steps provided the power level is limited to mitigate the increase in local peaking. The staff agrees that at power levels of 50% or below, RPI indications of ±24 steps are acceptable. For power levels above 50% the present misalignment limits will remain in effect.

2.3 On-Line Calibration of the RPI Channels

Calibration of the RPI channels is currently performed with the reactor at hot zero power. To assess the impact of on-line calibration of RCCAs on core safety limits, the NODE-P2 model was applied to cycles 12 and 13. Individual RCCAs were inserted in two node increments (24 axial nodes modeled in NODE-P2) to 0 steps withdrawn followed by withdrawal to 225 steps. At each insertion step, the core peaking factors, axial flux difference and quadrant tilt were calculated. The insertion/withdrawal of each of the 53 RCCAs was simulated at BOC, MOC and EOC.

Results were presented for the "worst case" RCCAs. A "worst case" RCCA is one which has a high reactivity worth resulting in the largest increase in local peaking factors and/or the greatest effect on global core power distributions. The results of the analysis lead to the following conclusions:

- The limiting core parameter with respect to on-line calibration of RCCAs is core quadrant tilt.
- At 30% power the TS limits on F_q and F delta H are met.
- RCCA calibration has some effect on core axial flux difference. For some cases the operating band is exceeded, the envelope is not. (Operation outside the band but within the envelope is permitted for 1 hour in 24 hours.)

It is anticipated that RPI calibration will require less than 2 hours. By conducting on-line calibration at a nominal 30% power level (not to exceed 35%) the quadrant tilt and axial flux difference TS limits should be met. The current TSs on quadrant tilt and axial flux difference will remain in effect during on-line calibration of the RPI channels and thus will stop the on-line calibration should these TS limits be approached.

If necessary, the first on-line calibration of RPIs would be performed during startup of the cycle at the 30% hold point, where a incore flux map is taken. The results of this map provide further verification that all control rods are in alignment with their bank demand position.

2.4 Validation/Verification of the NODE-P2 Code

The NODE-P2 code was used for the analysis of the extension of the RPI deviations and the on-line calibration. This code is part of the ARMP-02 documentation submitted in EPRI NP-4574-CCM. While NRC has accepted the NODE-P2, Con Edison had not previously submitted any benchmarking justifying their use of the code. During this review, the NRC staff requested this documentation.

Con Edison supplied data from cycles 10-13. It consisted of comparisons of measured vs predicted as well as Con Edison predictions using NODE-P2 with predictions by their fuel vender using another code and some data comparing measured to both sets of predictions.

There were two problems with the benchmarking provided. First, the comparisons were fairly good, but not as good as would be expected. In some cases, particularly the initial boron concentration the deviation between Con Edison's predictions and the measured data and the deviation between Con Edison's predictions and the vender's predictions has gotten larger with each succeeding cycle. The deviation between the Con Edison's predictions and the measurement for cycle 12 was about twice that usually observed throughout the industry at the present time. The deviation between the Con Edison and vendor predictions for Cycle 13, the upcoming cycle, is also twice that usually observed. The second problem was that almost all the benchmarking was calculations involving a symmetric core. Whereas the misalignment and on-line calibration calculations involved asymmetric calculations, which are more difficult.

Con Edison's analysis had concluded that the ± 24 step misalignment was acceptable up to 85% power and that the on-line RPI calibration was acceptable up to 50% power. The NRC staff's review of the analysis and benchmarking did not find sufficient justification for these power levels. However, the margin gained in going to 50% for the ± 24 step misalignment and a nominal 30% for the on-line calibration would be sufficient to justify these lower power levels for Cycle 13.

2.5 Fuel Thermal/mechanical Duty During On-line RPI Calibration

On-line calibration of the RPI channels will require the insertion of RCCAs from the fully withdrawn position to the fully inserted position followed by subsequent RCCA withdrawal. This will be carried out over the time periods of a few minutes and from a reduced power level. As individual RCCAs are inserted, the core power distribution is shifted away from the RCCA causing a power peak in the diametrically opposed core octant. Thus, a few fuel rods may be subjected to a mild power cycle during the calibration exercise. This, however, is minimized by allowing the core power to drift downward as the RCCA is inserted and back up as the RCCA is withdrawn.

In order to assess the effect of such power cycling on the fuel rod thermal mechanical duty, a FRAPCON model of IP2 fuel was developed. The limiting rods which experience the greatest power cycle during on-line calibration, were identified by examining the NODE-P2 simulations. The limiting rod was defined as the rod which experienced the largest nodal power increase during on-line calibration exercises. This fuel rod was then evaluated with respect to fuel thermal mechanical duty and the following conclusions were reached.

- In the fuel rod subjected to the highest linear heat generation rate during calibration, the resulting power cycle is mild and the effect on fuel thermal/mechanical duty is insignificant.
- Some fuel rods in the assembly receiving the RCCA are subjected to a relatively large power cycle with no significant calculated effect on fuel rod thermal mechanical duty.
- There is no calculated increase in fission gas release due to the calibration exercise which would lead to an increase potential for stress corrosion cracking of the cladding.
- Clad stress levels during calibration are not increased significantly over the steady state values just prior to calibration.

The NRC staff has reviewed the data provided from the analysis and agrees with Con Edison's conclusions.

2.6 Validation and Verification of FRAPCON2/V1MOD5

FRAPCON2/V1MOD5 is the most recent in the FRAPCON series of fuel rod response modeling programs. This program was developed by EG&G Idaho Inc., and Pacific Northwest Laboratory. Northeast Technology Corporation (NETCO), a Con Edison contractor, has previously validated and verified the mainframe version of FRAPCON by simulating the power exposure histories of a number of commercial LWR fuel rods as well as instrumented rods irradiated at the Halden Test Reactor. A summary of that work was provided.

The mainframe program was converted to execute on a microcomputer with 486 processor. This microcomputer version was validated and verified by simulating the same matrix of test rods as used for the validation and verification of the mainframe version. Details of the major conclusions on clad creep deformation, pellet relocation, fission gas release, fuel stack length and pellet density, fuel clad axial deformation, fuel temperatures, and ZrO_2 film were provided. Con Edison concluded that the FRAPCON2/V1MOD5 predictions with measured LWR fuel performance data serves to confirm that the code is providing accurate predictions of the fuel rod parameters key to reliable fuel performance. The NRC staff agrees with this conclusion.

2.7 Impact of On-Line Calibration on Postulated Plant Transients and Accidents

All postulated plant transients and accidents assessed in Section 14 of the IP2 FSAR have been reviewed and evaluated. This review was based on the assumption that each of the transients and accidents was initiated during the calibration of the RCCA at the point of RCCA insertion at which peaking factors are at their maximum values. The power level was 50% or rated or less and the reduced power would generally be expected to mitigate any increase in core peaking. The results of this analysis were presented in table form listing the event and the consequences. In all cases the consequences are no more severe than the bounding analyses documented in the FSAR.

3.0 SUMMARY

The analyses and evaluations completed for IP2 for fuel cycles 11, 12, and 13 demonstrate that the RPI deviation band can be extended to ± 24 steps for low power. As discussed in section 2.4, the validation /verification of the NODE-P2 code is not sufficient to justify the extended deviation band to the 85% power level,as initially proposed by Con Edison, for long-term operation. However, the benchmarking provided, together with the increase in allowable peaking factors for lower power and the core monitoring of quadrant tilt and axial flux difference by the ex-core dectors, is sufficient to allow the RPI deviation band to be extended to ± 24 steps for power levels below 50% for cycle 13 only.

Similarly, the NRC staff does not find sufficient justification for the 50% power level initially proposed by Con Edison for on-line calibration. The nominal power level of 30% (not to exceed 35%) is acceptable for on-line calibration of the RPI channels for cycle 13 only.

Therefore, the revised Con Edison request for changes to TS 3.10 to reflect extended RPI deviation limits of ± 24 steps for up to 50% power and on-line calibration of the RPI channels at a nominal power level of 30% (not to exceed 35%) is acceptable for cycle 13 only. The NRC staff also finds the update of the TS Bases, to reflect these changes, acceptable.

In order to justify the ± 24 step misalignment for the 85% power level and/or allow the on-line calibration at the 50% power level for future cycles, as initially requested, the NRC staff requires further benchmarking including comparisons of NODE-P2 calculations with calculations by the vendor or with measurements for asymmetric cases as well as comparisons with the measured data for cycle 13. Discrepancies between NODE-P2 predictions and other predictions and between NODE-P2 predictions and measurements must be adequately explained.

It should be further noted that the calculated peaking factors for the ± 24 step misalignment and the on-line calibration varied significantly between cycles 11, 12, and 13. Thus, prior to startup of each future cycle, these calculations would need to be repeated to ensure that the peaking factors were being maintained.

4.0 STATE CONSULTATION

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.

5.0 ENVIRONMENTAL CONSIDERATION

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 (59 FR 37069). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). 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.

6.0 CONCLUSION

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: Margaret Chatterton

Date: April 28, 1995

Stephen E. Quinn
Consolidated Edison Company
of New York, Inc.

Indian Point Nuclear Generating
Station Units 1/2

cc:

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, NY 12223

Mr. Charles W. Jackson
Manager of Nuclear Safety and
Licensing
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, NY 10511

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P.O. Box 38
Buchanan, NY 10511

Mr. Brent L. Brandenburg
Assistant General Counsel
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, NY 10003

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Mr. Peter Kokolakis, Director
Nuclear Licensing
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

Mr. Walter Stein
Secretary - NFSC
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, NY 10003

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406