ATTACHMENT A

PROPOSED TECHNICAL SPECIFICATIONS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
JUNE, 1994

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 following limits:

(a) $F^N_{\Delta H} \leq 1.62 \, [1 + 0.3 \, (1-P)]$

(b) For ≤ 25% steam generator tube plugging:

$$F_Q(Z) \leq (2.32/P) \times K(Z) \text{ for } P > .5$$

$$F_Q(Z) \leq (4.64) \times K(Z) \text{ for } P \leq .5$$

Where P is the fraction of full power at which the core is operating; K(Z) is the fraction given in Figure 3.10-2 and Z is the core height location of F_Q.

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 this specification 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 under Item 3.10.2.1, 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 a ± 5% band 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 ± 5% target band for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference does not exceed an envelope bounded by -11 percent and +11 percent at 90% power and increasing by -1 percent and +1 percent 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 Rod Insertion Limits

3.10.4.1 The shutdown rods shall be fully withdrawn 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 shown in Figure 3.10-3.

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). 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 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) 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.

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) of Specification 3.10.2.1 times the normalized peaking factor axial dependence of Figure 3.10-2 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 the normalized envelope of Figure 3.10-2 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 $F^N_{\Delta H} \leq 1.62/1.08$. 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) 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 Specification 3.10.2, 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 Figure 3.10-2 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 deviation of ±5 percent ΔI are 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. Figure 3.10-5 shows a typical construction of the target flux difference band at BOL and Figure 3.10-6 shows the typical variation of the full-power value with burnup.

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 +14 to -14 percent (+11 percent to -11 percent indicated) increasing by ± 1 percent for each 2 percent decrease in rated power. 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 ±5 percent band 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) 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 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. Below the rating specified in the Core Operating Limits Report (COLR), the accuracy can be relaxed to ≤24 steps because the power peaking factor limits would not be exceeded for any indicated misalignment within this band. 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.

Reference

1. UFSAR Section 14.3

ATTACHMENT B

SAFETY ASSESSMENT

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
JUNE, 1994

Section I - Description of Changes

This application for amendment to the Indian Point Unit 2 Technical Specifications seeks to amend 1) Section 3.10.6.1 to allow extended RPI deviation limits and 2) Section 3.10.4.4 to allow on-line calibration of the RPIs. In addition, it is proposed that the Basis for Section 3.10 be changed to reflect the above. Also, we propose that Section 3.10.6.2 be changed to clarify the operability requirements during calibration.

Section II - Evaluation of Changes

An evaluation of the changes is provided in Attachment C. Proposed additions to the Core Operating Limit Report (COLR) are contained in Attachment E.

Section III - No Significant Hazards Evaluation

Consistent with the requirements of 10 CFR 50.92, the enclosed application involves no significant hazards based on the following information which is detailed in Attachment C.

1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response:

Neither the probability nor the consequences of an accident previously analyzed is increased due to the proposed changes. All peaking factors will remain within the limits of the Technical Specifications. Both the shutdown margin and the axial flux difference will be maintained within the limits of the Technical Specifications. There will be no fuel damage due to the changes. All design and safety criteria will be met.

2) Does the proposed license amendment create the possibility of a new or different kind of accident from any previously evaluated?

Response:

The changes will not create the possibility of a new or different kind of accident. The calibration will be performed using plant procedures that have been reviewed and approved by Con Edison's Safety Committees. It has been shown that even with the new RPI deviation bands and on-line calibration, all power distribution limits will be met.

3) Does the proposed amendment involve a significant reduction in the margin of safety?

Response:

The proposed amendment does not involve a significant reduction in the margin of safety. There will be no change in the power distribution limits used in the design and safety analyses and the required shutdown margin will be maintained. It has been shown that there is no fuel failure as a result of this change.

Section IV - Impact of Changes

These changes will not adversely impact the following:

The ALARA Program, since it will not increase the amount of radioactive material nor does it involve the handling of radioactive material.

The Security Program, since it only involves Control Room operating procedures for instrument calibration that do not affect security.

The Fire Protection Program, since it only involves changes in operating procedure for instrument calibration that do not involve fire protection equipment nor does it increase the risk of fire.

The Emergency Plan since the margin of safety is maintained and hence there is no adverse affect on the consequences of accidents.

The FSAR and SER conclusions since the margin of safety is maintained.

Overall Plant Operation since it involves changes in procedures for instrument calibration and will not affect safe operation of the plant.

The Environment since it will not increase any releases to the environment.

Section V - Conclusion

These changes will be implemented by plant operating procedures which will be reviewed by Con Edison's Safety Committees to ensure that they will not affect the safe operation of the plant. The changes will not result in core physics parameters (peaking factors, shutdown margins, axial flux difference) exceeding the Technical Specification limits. No increase in fuel damage is expected. All design and safety criteria will be met. Therefore, the incorporation of these changes will not increase the probability or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the safety analysis report.

These changes only involve changes in the allowable band and calibration of existing instrumentation that as detailed above meet all design and safety criteria and Technical Specification limits. No new instrumentation is used nor is the plant operated in a new manner. Therefore, the incorporation of these changes will not create the possibility for an accident or malfunction of a new or different kind from any evaluated previously in the Safety Analysis Report.

These changes will not result in any physics parameter exceeding its Technical Specification limit. These changes will not affect the conclusions of any accident analysis. Therefore, the incorporation of these changes will not reduce the margin of safety as defined in the bases for any Technical Specification.

Therefore, based on the above, the incorporation of the proposed changes does not constitute an unreviewed safety question and involves no significant hazards considerations as defined in 10 CFR 50.92.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Nuclear Facilities Safety Committee (NFSC). Both Committees concur with the above.