ATTACHMENT A

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Consolidated Edison Company of New York, Inc.

Power Authority of the State of New York

Indian Point Unit No. 3

Docket No. 50-286

April 18, 1977

LIST OF FIGURES

where

ΔT_o = Indicated ΔT at rated power, °F

T_{avg} = Average temperature, °F

$T' \leq (571.5)$°F

P = Pressurizer pressure, psig

P' = Indicated nominal pressurizer pressure at rated power = 2235 psig

Four Loop Operation	Three Loop Operation
$K_1 \leq 1.200$	$K_1 \leq 1.110$
$K_2 \geq 0.0129$	$K_2 \geq 0.0129$
$K_3 \leq 0.00073$	$K_3 \leq 0.00073$

K_1 is a constant which defines the over temperature ΔT trip margin during steady state operation if the temperature, pressure and $f(\Delta I)$ terms are zero.

K_2 is a constant which defines the dependence of the overtemperature ΔT set point to T_{avg}.

K_3 is a constant which defines the dependence of the overtemperature ΔT set point to pressurizer pressure.

ΔI = $q_t - q_b$, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power.

$f(\Delta I)$ = a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are as defined above such that:

(a) for $q_t - q_b$ within -20, +16 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +16 percent, the ΔT trip set point shall be automatically reduced by an equivalent of 6.0 percent of rated power.

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -20 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 1.5 percent of rated power.

(5) Overpower ΔT

$$\Delta T \leq \Delta T_o \, [K_4 - K_5 \frac{dT_{avg}}{dt} - K_6 (T_{avg} - T'') - f(\Delta I)]$$

where

ΔT_o = indicated ΔT at rated power, °F

T_{avg} = average temperature, °F

T'' = indicated T_{avg} at nominal conditions at rated power, $\leq$(571.5)°F

K_4 $\leq$ 1.09

K_5 = 0 for decreasing average temperature

$\geq$ 0.175 sec/°F for increasing average temperature

K_6 = 0 for $T \leq T''$

$\geq$ 0.00127 for $T > T''$

K_4 is a constant which defines the overpower ΔT trip margin during steady state operation if the temperature and the $f(\Delta I)$ terms are zero.

K_5 is a constant determined by dynamic considerations to compensate for piping delays from the core to the loop temperature detectors; it represents the combination of the equipment static gain setting and the time constant setting.

K_6 is a constant which defines the dependence of the overpower ΔT setpoint to T_{avg}.

$f(\Delta I)$ = as defined above.

$\frac{dT_{avg}}{dt}$ = rate of change of T_{avg}

(6) Low reactor coolant loop flow:

(a) $\geq$ 90% of normal indicated loop flow

(b) Low reactor coolant pump frequency - $\geq$ 55.0 cps

(7) Undervoltage - $\geq$ 70% of normal voltage

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:

To ensure:

1. Core subcriticality after reactor trip,
2. 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. 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 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:

$$F_Q(Z) \leq (2.32/P) \times K(Z) \text{ for } P > 0.5$$
$$F_Q(Z) \leq (4.64) \times K(Z) \text{ for } P \leq 0.5$$
$$F^N_{\Delta H} \leq 1.55\ [1 + 0.2\ (1-P)]\ \left(1 - \frac{F^N_{\Delta H} \text{ Penalty } (\%)}{100}\right)$$

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. The $F^N_{\Delta H}$ Penalty, as a function of region average burnup, is given in Figure 3.10-2a, and as further modified by specification 3.10.2.2.2.

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 If the measured total peaking factor, F_Q^{Meas}, after having been increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement uncertainty exceeds the 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 power equal to the ratio of the F_Q limit to measured value corrected for manufacturing tolerance and measurement uncertainty. If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the total peaking factor limit specified under Item 3.10.2.1 is met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing.

3.10.2.2.2 If the measured enthalpy rise hot channel factor $F_{\Delta H}^{N}$, after having been increased by four percent to account for measurement uncertainty exceeds the limit specified under Item 3.10.2.1, then, within eight hours verify that sufficient margins (excess primary coolant flow and/or reduce T' in the overtemperature Δ T trip and/or a lower core inlet fluid temperature, in conjunction with a lower value of T') exist to assure DNBR greater than or equal to 1.30. However at no time can the margins exceed the $F_{\Delta H}^{N}$ penalty. Verification of the margins due to primary coolant flow rate and/or core inlet fluid temperature shall be continued, once every seven days, until such time as

Incore mapping verifies that the $F_{\Delta H}^{N}$ hot channel factor limit specified under Item 3.10.2.1 is met.

3.10.2.2.3 If sufficient margins can not be demonstrated to fully compensate for the $F_{\Delta H}^{N}$ penalty within the time frame specified in 3.10.2.2.2 then (a), the reactor power and high neutron flux trip setpoint shall be reduced, within an additional 2 hours, so as not to exceed a fraction of rated power equal to the ratio of the $F_{\Delta H}^{N}$ limit, corrected for available margins, to the measured value corrected for measurement uncertainty and (b) if subsequent incore mapping cannot, within an additional 24-hour period, demonstrate that the enthalpy rise hot channel factor, $F_{\Delta H}^{N}$, as modified by available margins specified in 3.10.2.2.2 is 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 for each excore channel as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target flux differences must be updated each effective full power month by linear interpolation using the most recent measured value and a value of 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 of all but one operable excore channel shall be maintained within a $\pm$ 5% band about the target flux difference.

F_Q^E, 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.

$F_{\Delta H}^N$, 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 $F_{\Delta H}^N$ 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 $F_{\Delta H}^N$.

An upper bound envelope of 2.32 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 this upper bound normalized envelope of Figure 3.10-2 demonstrate a peak clad temperature below the 2200°F limit.[2]

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 moveable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F_{\Delta H}^N$ there is a 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g. rod misalignment) affect $F_{\Delta H}^{N}$, 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 $F_{\Delta H}^{N}$ 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 $F_{\Delta H}^{N}$ is less readily available. When a measurement of $F_{\Delta H}^{N}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system.

The $F_{\Delta H}^{H}$ limit (Specification 3.10.2.1) includes the effect of fuel rod bowing. This bowing has been observed in Westinghouse LOPAR fuel, and exceeds the design value assumed in the pitch reduction. Conservative models [3] have been developed to account for the effect of fuel rod bow on DNB. These models in conjunction with the generic DNB margins appropriate for Indian Point Unit No. 3 result in the $F_{\Delta H}^{N}$ penalty versus burnup given in Figure 3.10-2a.

Additional margin to DNBR of 1.30 include excess primary coolant flow above that used in the safety analyses. The measured primary coolant flow, if used to verify margins to DNBR of 1.30, shall be decreased by two percent for measurement uncertainty when using secondary-side calorimetrics and vessel ΔT, and 3.5 percent when using elbow tap measurements. The excess primary coolant flow is the difference between the measured primay coolant flow including measurement uncertainty and the design primary coolant flow. The relationship between primary coolant flow and $F_{\Delta H}^{N}$ margin is given by,

$$\%\text{Excess Primary Coolant Flow} \times 0.57 = \%\ F_{\Delta H}^{N}\ \text{Margin}$$

A reduction in the T' value in the overtemperature ΔT setpoint, can be used to obtain additional DNB margin. However if no actual reduction in the core inlet fluid temperature (T_{IN}) is present, the minimum allowable value of T' (indicated Tavg at nominal conditions at rated power) is 566.1 °F, and the $\%F_{\Delta H}^{N}$ margin is given by,

$$(571.5^{o}F - T') \times 0.57 = \%\ F_{\Delta H}^{N}\ \text{margin.}$$

The 566.1 °F lower limit for T', for no change to T_{IN}, ensures that the limiting DNB case, for which the overtemperature ΔT setpoint is not used, remains above DNBR = 1.30.

If an additional reduction in T' is to be made below 566.1^{o} F, the measured value of the core inlet fluid temperature shall be verified to be lower than the design value by 1^{o} F for every 1^{o} F further reduction in T'. The measured value T_{IN} shall be increased by 1^{o} F for measurement uncertainty.

In no event shall the margins, due to the increased primary coolant flow and reduced T', exceed the $F^{N}_{\Delta H}$ penalty as given in Figure 3.10-2a.

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 basis including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomolies 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 13 steps precludes a rod misalignment no greater than 15 inches with consideration of maximum instrumentation error.

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

3. The full length and part length control bank insertion limits are not violated.

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 inches away from its demand position. An indicated misalignment less than 13 steps does not exceed the power peaking factor limits. 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 13 step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

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

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

REFERENCE

1. WCAP-8576, "Augmented Startup and Cycle 1 Physics Program", August 1975
2. FSAR Appendix 14C
3. NRC Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors, Revision I, February 16, 1977.

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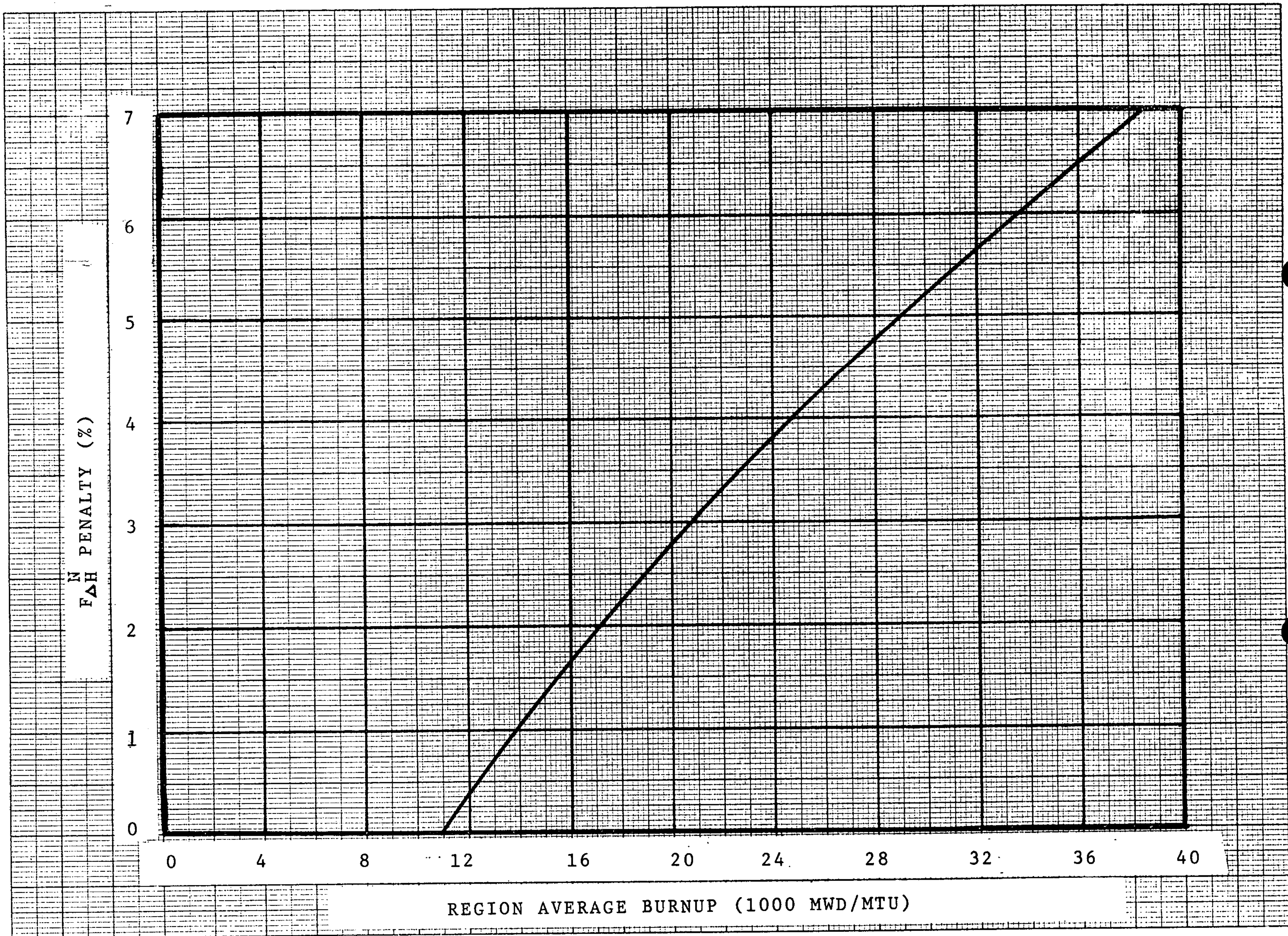

Figure 3.10-2a Fuel Rod Bow $F_{\Delta H}^{N}$ Penalty Versus Region Average Burnup

ATTACHMENT B

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Safety Evaluation

Consolidated Edison Company of New York, Inc.

Power Authority of the State of New York

Indian Point Unit No. 3

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SAFETY EVALUATION

The proposed changes to the Indian Point Unit No. 3 Technical Specifications, contained in Attachment A to this Application, would add the requirement of an additional $F_{\Delta H}^{N}$ (enthalpy rise hot channel factor) penalty, as a function of region average burnup, and credit for measurable DNB margins to offset the $F_{\Delta H}^{N}$ penalty. These requirements would ensure that the margins to departure from nucleate boiling identified in the NRC Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors, Revision 1, February 16, 1977 are maintained.

To achieve a greater degree of conformity with the Standard Technical Specifications, certain editorial corrections are also proposed. These changes would eliminate areas of possible ambiquity.

The proposed changes do not in any way alter the safety analyses performed for Indian Point Unit No. 3. The proposed changes have been reviewed by Con Edison's Station Nuclear Safety Committee and Nuclear Facilities Safety Committee. Both committees concur that these changes do not represent a significant hazards consideration and will not cause and change in the types of or increase in the amounts of effluents or any change in the authorized power level of the facility.