ATTACHMENT A

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Technical Specification
Page Revisions

Consolidated Edison Company of New York, Inc.
Power Authority of the State of New York

Indian Point Unit No. 3
Docket No. 50-286
Facility Operating License No. DPR-64

December, 1977

LIST OF FIGURES

AMENDMENT No.

Rod withdrawal block and load runback occurs if reactor trip setpoints are approached within a fixed limit.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in a DNBR of less than 1.30.[4]

References

(1) FSAR Section 3.2.2

(3) WCAP-8147, "Fuel Densification-Indian Point Nuclear Generating Unit No. 3", July, 1973; Westinghouse Non-Proprietary Class 3.

(4) FSAR Section 14.1.1

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. 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.

3.10.4 <u>Rod Insertion Limits</u>

3.10.4.1 The shutdown rods shall be fully withdrawn when the reactor is critical or approaching criticality (i.e., the reactor is no longer subcritical by an amount equal to or greater than the shutdown margin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 3.10-4 or Figure 3.10-5.

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

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3.10.5 Rod Misalignment Limitations

3.10.5.1 If a control rod is misaligned from its bank demand position by more than 13 steps, 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 requirements 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 If a rod position indicator channel is out of service then:

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

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

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

3.10.6.3 If a control rod having a rod position indicator channel out of service, is found to be misaligned from 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 fails to meet the requirements of 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 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry.

(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 moveable 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 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 **control rod** bank insertion limits are not violated.

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

The permitted relaxation in $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In 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 F_Q upper bound envelope of 2.32 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 the 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

-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 lower, 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 FSAR Section 14.1 events, the core is protected from overpower and a minimum DNBR of 1.30 by an automatic protection system. Compliance with operating procedures is assumed as a precondition for FSAR Section 14.1 events. 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 initiated. Therefore, the Technical 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

Figure 3.10-3

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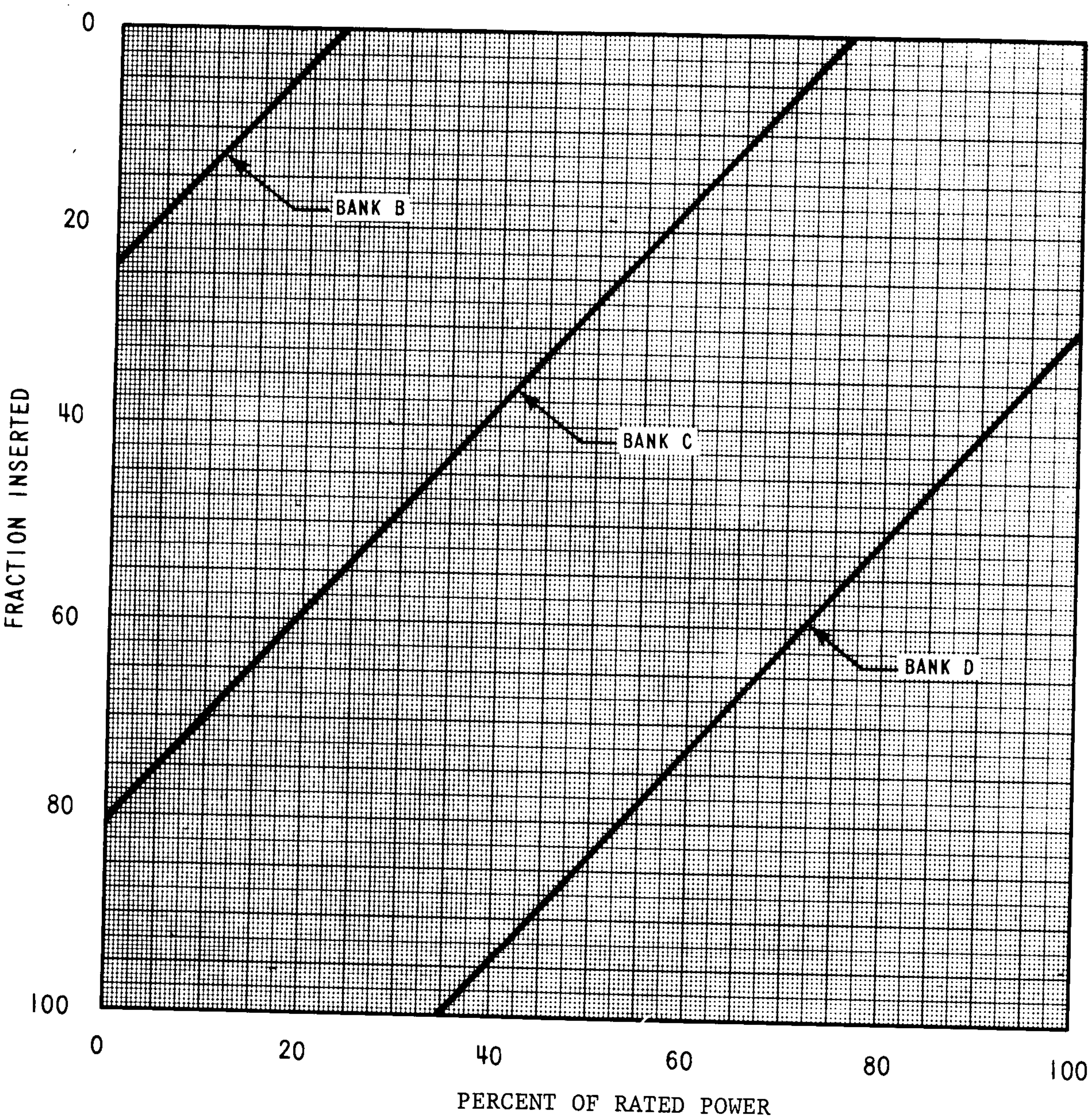
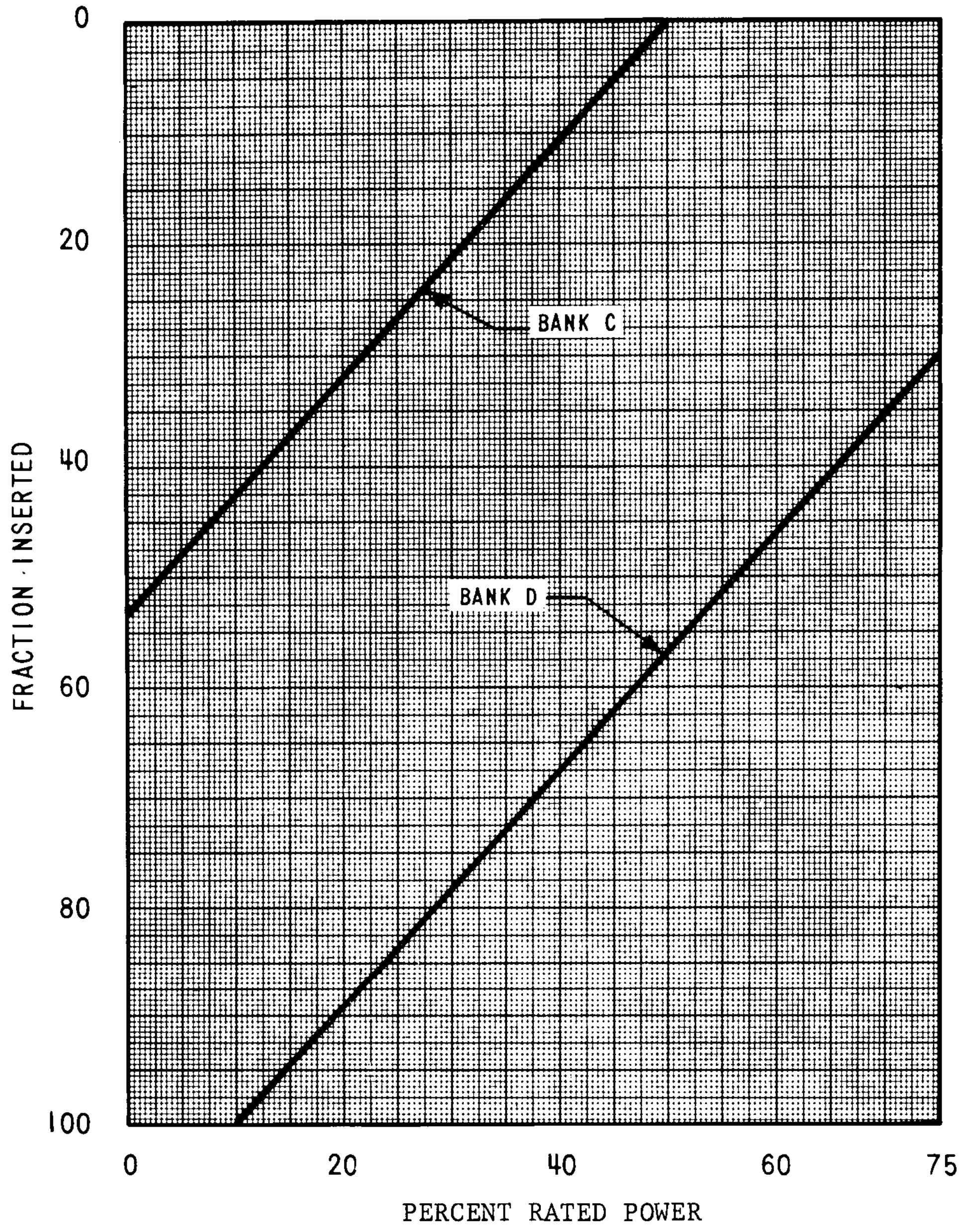


NOTE: BANK A IS FULLY WITHDRAWN AT ZERO POWER

Figure 3.10-4. Insertion Limits 100 Step Overlap Four Loop Operation



NOTE: BANKS A & B ARE FULLY WITHDRAWN AT ZERO POWER

Figure 3.10-5. Insertion Limits 100 Step Overlap 3 Loop Operation

TABLE 4.1-3 (Sheet 1 of 1)

FREQUENCIES FOR EQUIPMENT TESTS

		Check	Frequency
1.	Control Rods	Rod drop times of all control rods	R
2.	Control Rods	Partial movement of all control rods	Every 2 weeks during reactor critical operations
3.	Pressurizer Safety Valves	Set point	R
4.	Main Steam Safety Valves	Set point	R
5.	Containment Isolation System	Automatic actuation	R
6.	Refueling System Interlocks	Functioning	Prior to each refueling outage
7.	Primary System Leakage	Evaluate	5 days/week
8.	Diesel Fuel Supply	Fuel Inventory	Weekly
9.	Turbine Steam Stop, Control Valves	Closure	Monthly
10.	L.P. Steam Dump System (6 Lines)	Closure	Monthly
11.	Service Water System	Each pump starts and operates for 15 minutes (unless already operating)	Monthly
12.	City Water Connections to Charging Pumps and Boric Acid Piping	Temporary connections available and valves operable	R
13.	RHR Valves 730 and 731	Automatic isolation and interlock action	R*

R Each refueling outage
* If not done during the previous 18 months, the check will be performed the next time the plant is cooled down.

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5. There are 53 control rods in the reactor core. The control rods contain 142 inch lengths of silver-indium-cadmium alloy clad with the stainless steel.[5]

Reactor Coolant System

1. The design of the reactor coolant system complies with the code requirements.[6]

2. All piping, components and supporting structures of the reactor coolant system are designed to Class I requirements, and have been designed to withstand the maximum potential seismic ground acceleration, 0.15g, acting in the horizontal and 0.10g acting in the vertical planes simultaneously with no loss of function.

3. The total liquid volume of the reactor coolant system, at rated operating conditions, is 11,522 cubic feet.

References

(1) FSAR Section 3.2.2
(2) FSAR Section 3.2.1
(3) FSAR Section 3.2.1
(4) FSAR Section 3.2.3
(5) FSAR Sections 3.2.1 & 3.2.3
(6) FSAR Table 4.1- 10

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
Docket No. 50-286
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Safety Evaluation

This Application is submitted to facilitate the removal of the part length control rods from the reactor vessel. Technical Specification 3.10.4.5 requires that these part length control rods be withdrawn and excluded from the core except during low power physics tests and for axial offset calibration tests performed below 75% of rated power. This restriction was intended to assure that the axial power shape used a basis for the DNB analyses in the Safety Analysis Report was met. As a consequence, the part length control rods are neither used for shutdown margins nor used for control of axial power distributions. Instead, the full length control rods alone perform these functions.

The part length control rods are no longer required for axial offset calibration tests or during low power physics tests. It was consequently determined that the system design could be improved and the chances of a misaligned part length control rod could be eliminated by removing these control rods from the reactor vessel. The proposed changes would delete all description and mention of the part length control rods from the technical specifications and its bases.

The removal of the part length control rods from the vessel will not cause any safety problem or impair the safe operation of the reactor in any way. A vibration analysis was conducted for the leadscrew rod assembly. This study indicated that there will be no unacceptable level of vibration as a result of part length control rod removal, and the leadscrew rod assembly would not be subjected to any appreciable additional stress or fatigue problems due to flow-induced vibration. Reactor coolant flow from the core to the upper internals will be essentially the same because inserts will be placed in the fuel assembly thimble tubes where part length rods have been removed. Upper plenum coolant temperatures and reactor head temperature will also be unchanged because the leadscrew shafts will be left in place and flow resistance will not be changed.

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 the Station Nuclear Safety Committee and the Nuclear Facilities Safety Committee. Both committees concur that these changes do not represent a significant hazards consideration and will not cause any change in the types or increase in the amounts of effluents or any change in the authorized power level of the facility.