June 17, 1998

Mr. James Knubel
Chief Nuclear Officer
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M99085)

Dear Mr. Knubel:

The Commission has issued the enclosed Amendment No. 180 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3 (IP3). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated June 25, 1997, as supplemented by letter dated June 2, 1998. The amendment changes the TSs to allow for up to +17/-12 steps of control rod misalignment when power is greater than 85%.

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:

George F. Wunder, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures: 1. Amendment No. 180 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\IP3\99085.AMD

*See previous concurrence

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

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Docket No. 50-286

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2. Safety Evaluation

cc w/encls: See next page

James Knubel
Power Authority of the State
of New York

Indian Point Nuclear Generating
Unit No. 3

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ISSUANCE OF AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-28

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 180
License No. DPR-64

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated June 25, 1997, as supplemented by letter dated June 2, 1998, 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-64 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 180, 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

S Bajwa

S. Singh Bajwa, 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: June 17, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 180

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.10-6	3.10-6
3.10-10	3.10-10
3.10-16	3.10-16

3.10.5 Rod Misalignment Limitations

3.10.5.1 At least once per shift (allowing one hour for thermal soak after rod motion) the position of each control or shutdown rod shall be determined:

a. For operation less than or equal to 85% of rated thermal power, the indicated misalignment between the group step counter demand position and the analog rod position indicator shall be less than or equal to 18 steps. A control or shutdown rod indicating a misalignment greater than 18 steps shall be realigned within one hour or the core peaking factors shall be determined within two hours and the requirements of Specification 3.10.2 applied.

b. For operation greater than 85% of rated thermal power, the indicated misalignment between the group step counter demand position and the analog rod position indicator shall be ±12 steps for less than or equal to 212 steps and +17, -12 steps for greater than 212 steps. A control or shutdown rod indicating a misalignment greater than the above mentioned steps shall be realigned within one hour or the core peaking factors shall be determined within two hours and the requirements of Specification 3.10.2 applied.

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.

(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, no additional allowances are necessary prior to comparison with the limit of section 3.10.2. A measurement uncertainty of 4% has been allowed for in determination of the design DNBR value.

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 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 group step counter demand position (operating at greater than 85% of rated thermal power with no accounting for peaking factor margin), or 18.75 inches (operating at less than or equal to 85% of rated thermal power). An indicated misalignment limit of 12 steps precludes a rod misalignment greater than 15 inches with consideration of instrumentation error and 18 steps indicated misalignment corresponds to 18.75 inches with 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.

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 infrequency 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 worth. 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.

The rod position indicator channel is sufficiently accurate to detect a rod ±7 inches away from its demand position. An indicated misalignment less than 12 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 moveable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 12 step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 5 day period is short compared with the time interval required to achieve a significant, non-uniform fuel depletion.

The assumed control rod drop time in the safety analysis is 2.7 seconds, consisting of 1.80 seconds for normal rod drop time plus additional margin which includes a seismic allowance. The required control rod drop time in Section 3.10.8 is therefore consistent with that assumed in the safety analysis.

REFERENCE

1. WCAP-8576, "Augmented Startup and Cycle 1 Physics Program," August 1975
2. FSAR Appendix 14C
3. Letter from J.P. Bayne to S.A. Varga dated April 23, 1985, entitled "Proposed Technical Specifications Regarding the Cycle 4/5 Refueling."
4. WCAP-14668, "Conditional Extension of the Rod Misalignment Technical Specification for Indian Point Unit 3," October 1996 (Proprietary).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 INTRODUCTION

By letter dated June 25, 1997, as supplemented by letter dated June 2, 1998, the Power Authority of the State of New York submitted an amendment to modify Indian Point Nuclear Generating Unit No. 3 Technical Specification (TS) 3.10.5.1, "Rod Misalignment Limitations." The proposed amendment would allow control rod misalignment of +17, -12 steps above 85% of rated thermal power (RTP) for greater than 212 steps. The June 2, 1998, supplement provided a clarification to the wording of the TS and did not change the staff's proposed finding of no significant hazards considerations.

The current TSs allow an indicated rod misalignment of ±18 steps when the core power is less than or equal to 85% of RTP. Above 85% RTP, the indicated misalignment between the group step counter demand position and the analog rod position indicator shall be less than or equal to 12 steps.

In many cases, indicated misalignments are inaccurate readings caused by fluctuations in the temperature of the control rod drive shafts. Such fluctuations can occur after rod control cluster assemblies (RCCAs) are withdrawn from the core during startup. When an indication of misalignment does occur, whether or not it is due to an inaccuracy, corrective actions per the TSs must be taken.

2.0 EVALUATION

The Analog Rod Position Indication (ARPI) system is designed to an accuracy of 12 steps. Therefore, in order to guarantee a rod misalignment of less than 24 steps (12 steps misalignment plus 12 steps ARPI uncertainty), the individual ARPI readings must be no larger than 12 steps. The justification for changing the allowed misalignment to +17, -12 steps for rod positions greater than 212 steps is the physical geometry of the core. To justify the increase in allowable rod misalignment, the following must be considered:

1) reactivity control
2) control rod misoperation (dropped rods and static rod misalignments)
3) rod ejection and
4) power operation with misaligned rods.

All of these depend upon the reactivity worth of the control rods. Since the difference in the reactivity worth is only a minute amount between control rod positions 224 and higher, any position above 224 has the same effect on the core. Thus, allowing a misalignment of +17 steps for positions of 212 steps withdrawn above 85% RTP is acceptable.

The staff has evaluated the effects of changing the allowed misalignment from + 12 steps to +17 steps for core power greater than 85% RTP at rod positions greater than 212 steps and found no adverse effects. Thus the proposed change to the Indian Point 3 Nuclear Power Plant Technical Specification 3.10.5.1, "Rod Misalignment Limitations" is acceptable.

3.0 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.

4.0 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 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent 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 (62 FR 45461. 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.

5.0 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: M. Chatterton

Date: June 17, 1998