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Docket No. 50-247 FEB 09 1981

Mr. John D. O'Toole
Assistant Vice President
Nuclear Affairs and Quality Assurance
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. O'Toole:

The Commission has issued the enclosed Amendment No. 66 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated May 21, 1980.

The amendment incorporates changes to the Technical Specifications to accommodate operation with low parasitic fuel.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 66 to DRR-26
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

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| DATE | 2/3/81 | 2/4/81 | 2/4/81 | 2/5/81 | 2/5/81 | | |

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Docket

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

February 9, 1981

Docket No. 50-247

Mr. John D. O'Toole
Assistant Vice President
Nuclear Affairs and Quality Assurance
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

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Sincerely,

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 66 to DPR-26
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

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

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.66
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated May 21, 1980, 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. 66 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 9, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 66

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
| 2.3-4 | 2.3-4 |
| 2.3-7 | 2.3-7 |
| Table 3-2 (second page) | Table 3-2 (second page) |
| Table 3-2 (third page) | Table 3-2 (third page) |
| 3.10-6 | 3.10-6 |
| 3.10-10 | 3.10-10 |
| 3.10-15 | 3.10-15 |
| Table 4.1-1 (third page) | Table 4.1-1 (third page) |
| 5.3-1 | 5.3-1 |

2. Protective instrumentation settings for reactor trip interlocks shall satisfy the following conditions:

   A. The reactor trips on low pressurizer pressure, high pressurizer level, and low reactor coolant flow for two or more loops shall be unblocked when:

      1) Power range nuclear flux $\geq$10% of rated power, or

      2) Turbine first stage pressure $\geq$10% of equivalent full load.

   B. The single loop loss of flow reactor trip may be bypassed when the power range nuclear instrumentation indicates $\leq$60% of rated power. The single loop loss of flow reactor trip may be bypassed below 75% of rated power only when the overtemperature $\Delta T$ trip setpoint has been adjusted to the three-loop operation value given in 2.3.1.B-4 above. The resetting of the overtemperature $\Delta T$ trip shall be performed by the I & C Repair Unit under the direct supervision of the Operations Staff of Consolidated Edison Company.

3. The Control Rod Protection System, for use when control rods are positioned in core locations containing LOPAR fuel, shall open the reactor trip breakers during RCS cooldown prior to $T_{cold}$ decreasing below 350°F.

Basis

The high flux reactor trips provide redundant protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis.(1)

The power range nuclear flux reactor trip high set point protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed set point, with allowance for errors, is consistent with the trip point assumed in the accident analysis. (2) (3)

The source and intermediate range reactor trips do not appear in the specification as these settings are not used in the transient and accident analysis (FSAR Section 14). Both trips provide protection during reactor startup. The former is set at about $10^{+5}$ counts/sec and the latter at a current proportional to approximately 25% of rated full power.

will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when only three loops are in operation.

The turbine and steam-feedwater flow mismatch trips do not appear in the specification as these settings are not used in the transient and accident analysis (FSAR Section 14).

To avoid mechanical interference due to thermal contraction between LOPAR fuel and the control rods (when control rods are positioned in core locations containing LOPAR fuel), an automatic backup to manual tripping of the control rods is provided. Prior to $T_{cold}$ decreasing below 350°F during RCS cooldown, the Control Rod Protection System will open the reactor trip breakers which unlatches the control rod drive shafts from the CRDMs.

<u>References</u>

(1) FSAR 14.1.1
(2) FSAR 14.1.2
(3) FSAR Table 7.4.2
(4) FSAR 14.3.1
(5) FSAR 14.1.2
(6) FSAR 7.2
(7) FSAR 3.2.1
(8) FSAR 14.1.6
(9) FSAR 14.1.9

TABLE 3-2 (Continued)

| | | 1 | 2 | 3 | 4 | 5 |
|---|---|---|---|---|---|---|
| 10. | Low Flow Loop ≥75% F.P. | 3/loop | 2/loop (any loop) | 2/operable loop | 1/operable loop | Maintain hot shutdown |
| | Low Flow Two Loops 10-75% F.P. | 3/loop | 2/loop (any two loops) | 2/operable loop | 1/operable loop | |
| 11. | Lo Lo Steam Generator Water Level | 3/loop | 2/loop | 2/loop | 1/loop | Maintain hot shutdown |
| 12. | Undervoltage 6.9 KV Bus | 1/bus | 2 | 3 | 2 | Maintain hot shutdown |
| 13. | Low frequency 6.9 KV Bus | 1/bus | 2 | 3 | 2 | Maintain hot shutdown *** |
| 14. | Quadrant power tilt monitors | 2 | NA | 1 | 0 | Log individual upper and lower ion chamber currents once/shift and after load change >10% |
| 15. | Turbine trip (overspeed protection) | 3 | 2 | 2 | 1 | Maintain hot shutdown |
| 16. | Control Rod Protection**** | 3 | 2 | 2 | 1 | During RCS cooldown, manually open reactor trip breakers prior to $T_{cold}$ decreasing below 350°F. Maintain reactor trip breakers open during RCS cooldown when $T_{cold}$ is less than 350°F. |

TABLE 3-2 (Continued)

| | |
|---|---|
| * | If two of four power channels greater than 10% F.P., channels are not required. |
| ** | If one of two intermediate range channels greater than $10^{-10}$ amps, channels are not required. |
| *** | 2/4 trips all four reactor coolant pumps. |
| **** | Required only when control rods are positioned in core locations containing LOPAR fuel. |
| F.P. = | Rated Power |

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 $\pm 12$ 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 within $\pm 12$ 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.

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.

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 basis 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 12 steps precludes a 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. 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 $P \leqslant 0.5$ (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

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 $\pm$7.5 inches away from its demand position for indicated control rod position less than or equal to 210 steps withdrawn. An indicated misalignment $\leq$12 steps does not exceed the power peaking factor limits. A misaligned rod of + 17 steps allows for an instrumentation error of 12 steps plus 5 steps that are not indicated due to the location relationship of the RPI coil stack and the control rod drive rod for indicated rod position greater than or equal to 211 steps withdrawn. The last five steps of rod travel are not indicated by the RPI because the drive rod and spider assembly have been raised three inches ($\sim$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. 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 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

TABLE 4.1-1 (CONTINUED)

| | Channel Description | Check | Calibrate | Test | Remarks |
|---|---|---|---|---|---|
| 24. | Turbine First Stage Pressure | S | R | M | |
| 25. | Logic Channel Testing | N.A. | N.A. | M | |
| 26. | Turbine Overspeed Protection Trip Channel (Electrical) | N.A. | R | M | |
| 27. | Control Room Ventilation | N.A. | N.A. | R | Check damper operation for accident mode with isolation signal |
| 28. | Control Rod Protection (for use with LOPAR fuel) | N.A. | R | * | |

* Within 31 days prior to entering a condition in which the Control Rod Protection System is required to be operable unless the reactor trip breakers are manually opened during RCS cooldown prior to $T_{cold}$ decreasing below 350°F and the breakers are maintained open during RCS cooldown when $T_{cold}$ is less than 350°F.

## 5.3 REACTOR

### Applicability

Applies to the reactor core, reactor coolant system, and emergency core cooling systems.

### Objective

To define those design features which are essential in providing for safe system operations.

A. Reactor Core

1. The reactor core contains approximately 87 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods.[(1)]

2. The average enrichment of the initial core is a nominal 2.8 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.3 weight per cent of U-235.[(2)]

3. The enrichment of reload fuel will be no more than 3.4 weight percent of U-235.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 66 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

Introduction

By letter dated May 21, 1980, Consolidated Edison Company of New York (the licensee) proposed changes to the Technical Specifications necessary to allow operation of Indian Point Unit No. 2 with low parasitic (LOPAR) fuel. The licensee intends to replace about one-third of the existing high parasitic (HIPAR) fuel with LOPAR fuel during the current refueling outage as the first step of a planned three-phase transition to an all LOPAR fuel core.

Evaluation

The Indian Point Unit No. 2 LOPAR fuel is essentially the same as the standard Westinghouse 15 x 15 LOPAR fuel used in many other plants, including Indian Point Unit No. 3. Based on our review of the licensee's May 21, 1980 letter and the October 7, 1980 response to our August 19, 1980 letter we conclude that the fuel's mechanical response to external loads should not change significantly. Tests were conducted which conclude that hydraulic compatibility exists between the LOPAR and HIPAR assemblies, and analyses were performed to demonstrate that the safety limits of the non-LOCA and LOCA accidents are satisfied. We, therefore, find the use of the LOPAR fuel in Indian Point Unit No. 2 to be acceptable.

The licensee proposed changes to the Technical Specifications in three areas to accommodate the change to LOPAR fuel.

Section 3.10 is being changed because the control rod spiders and drive rods will be raised three inches (approximately five steps) above the present HIPAR fuel in the fully withdrawn position when control rods are used in core locations containing LOPAR fuel. Thus, the rod position indication (RPI) system accuracy above 210 steps will be +17, -12 steps for LOPAR fuel instead of the ± 12 steps for HIPAR fuel.

Section 2.3 and Tables 3-2 and 4.1-1 are being changed because the licensee

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has designed a LOPAR fuel/control rod protection system to preclude fuel or control rod damage from the thermal contraction differences during cooldown. This system is required when control rods are used in core locations containing LOPAR fuel, and automatically opens the reactor trip breakers prior to the reactor coolant system cold leg temperature decreasing below 350°F.

Section 5.3.A.3 is being changed to delete the statement that "reload fuel will be similar in design to the initial core."

We have reviewed these changes to the Technical Specifications and find them acceptable.

## Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

## Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: February 9, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-247

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 66 to Facility Operating License No. DPR-26, issued to the Consolidated Edison Company of New York, Inc. (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 2 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of the date of issuance.

The amendment incorporates changes to the Technical Specifications to accommodate operation with low parasitic fuel.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commissions's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated May 21, 1980, (2) Amendment No.66 to License No. DPR-26, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the White Plains Public Library, 100 Martine Avenue, White Plains, New York. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 9th day of February, 1981

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing