Docket No.: 50-247

JAN. 8 1976

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Consolidated Edison Company
of New York, Inc.
ATTN: Mr. William J. Cahill, Jr.
Vice President
4 Irving Place
New York, New York 10003

Gentlemen:

The Commission has issued the enclosed Amendment No. 16 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 and is in response to your request dated January 6, 1976.

The amendment incorporates into the Indian Point Nuclear Generating Unit No. 2 Technical Specifications changes to the reporting requirements. The technical specifications are based on Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications," Revision 4.

We request that you use the formats presented in the Appendices to Regulatory Guide 1.16, Revision 4, for reporting operating information and that you report events of the type described under the section "Events of Potential Public Interest." Instructions for using these reporting formats are contained in Regulatory Guide 1.16 (a copy is enclosed for your use), and AEC report OOE-SS-001 titled "Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File (a copy of which was provided to you previously). This report is modified by updated instructions dated December 8, 1975, which are enclosed. Copy requirements are summarized in Regulatory Guide 10.1, "Compilation of Reporting Requirements for Persons Subject to NRC Regulations," a copy of which is also enclosed. This Guide will assist you in identifying reports that are required by the Commission's regulations set forth in Title 10 Code of Federal Regulations but are not contained in your technical specifications. Reports that are required by the regulations have not been repeated in your technical specifications.

Copies of the related Safety Evaluation and the Federal Register Notice also are enclosed.

OFFICE➤						
SURNAME➤						
DATE➤						

Please note that we have discontinued the use of separate identifying numbers for changes to technical specifications. Sequential amendment numbers will be continued as in the past.

Sincerely,

Original Signed By

Robert W. Reid, Chief
Operating Reactors Branch #4

Enclosures:
1. Amendment No. 16
2. Regulatory Guide 1.16,
   Revision 4
3. Updated Instructions
4. Regulatory Guide 10.1
5. Safety Evaluation
6. Federal Register Notice

cc: w/enclosures
See next page

OFFICE	ORB #4	ORB #4	C-ORB#4			
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DATE	1/1/75	1/8/76 ~~12/8~~/75	1/8/76 ~~12/~~ /75			

Consolidated Edison Company

cc: w/enclosures

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Albany, New YOrk 12210

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. 16
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 January 6, 1976, 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. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility 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, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

3. This license amendment becomes effective 30 days after the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Reactor Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: JAN. 8 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 16

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Replace the existing pages of the Technical Specifications listed below with the attached revised pages bearing the same numbers. Changes on these pages are shown by marginal lines.

Pages

ii, iii, and v

1-3 and 1-4

3.1-18

3.2-1

6-9 - 6-14

6-2

TA)F CONTENTS (Continued)

Amendment No. 16

TABLE OF CONTENTS (Continued)

Amendment No. 16

LIST OF FIGURES

1.6.1 Channel Check

A qualitative determination of acceptable operability by observation of channel behavior during operation. This determination shall include comparison of the channel with other independent channels measuring the same variable.

1.6.2 Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating action.

1.6.3 Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including alarm or trip, and shall be deemed to include the channel functional test.

1.7 Containment Integrity

Containment integrity is defined to exist when:

a. The required non-automatic containment isolation valves are closed and blind flanges are properly installed.

b. The equipment door is properly closed and sealed by the Weld Channel and Penetration Pressurization System.

c. At least one door in each personnel air lock is properly closed.

d. All automatic containment isolation valves are operable or closed.

e. The containment leakage satisfies Specification 4.4.

1.8 Reportable Occurrence

A Reportable Occurrence shall be any of those conditions specified in Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications" Sections C.2.a and C.2.b.

1.9 Quadrant Power Tilt

The quadrant power tilt is defined as the ratio of maximum to average of the upper excore detector currents or the lower excore detector currents whichever is greater. If one excore detector is out of service, the three in-service units are used in computing the average.

Amendment No. 16

Basis:

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of GPM may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks in the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation.

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Plant Operating Staff and will be documented in writing and approved by either the Plant Manager or his designated alternate. Under these conditions, an allowable primary system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also well within the capacity of one-charging pump and makeup would be available even under the loss of off-site power condition.

If leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the Chemical and Volume Control System.

Objective

To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

Specification

A. When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection.

B. The reactor shall not be made critical unless the following Chemical and Volume Control System conditions are met.

1. Two charging pumps shall be operable.

2. Two boric acid transfer pumps shall be operable.

3. The boric acid tanks together shall contain a minimum of 4400 gallons of 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) boric acid solution at a temperature of at least 145°F.

4. System piping and valves shall be operable to the extent of establishing one flow path from the boric acid tanks and one flow path from the refueling water storage tank to the Reactor Coolant System.

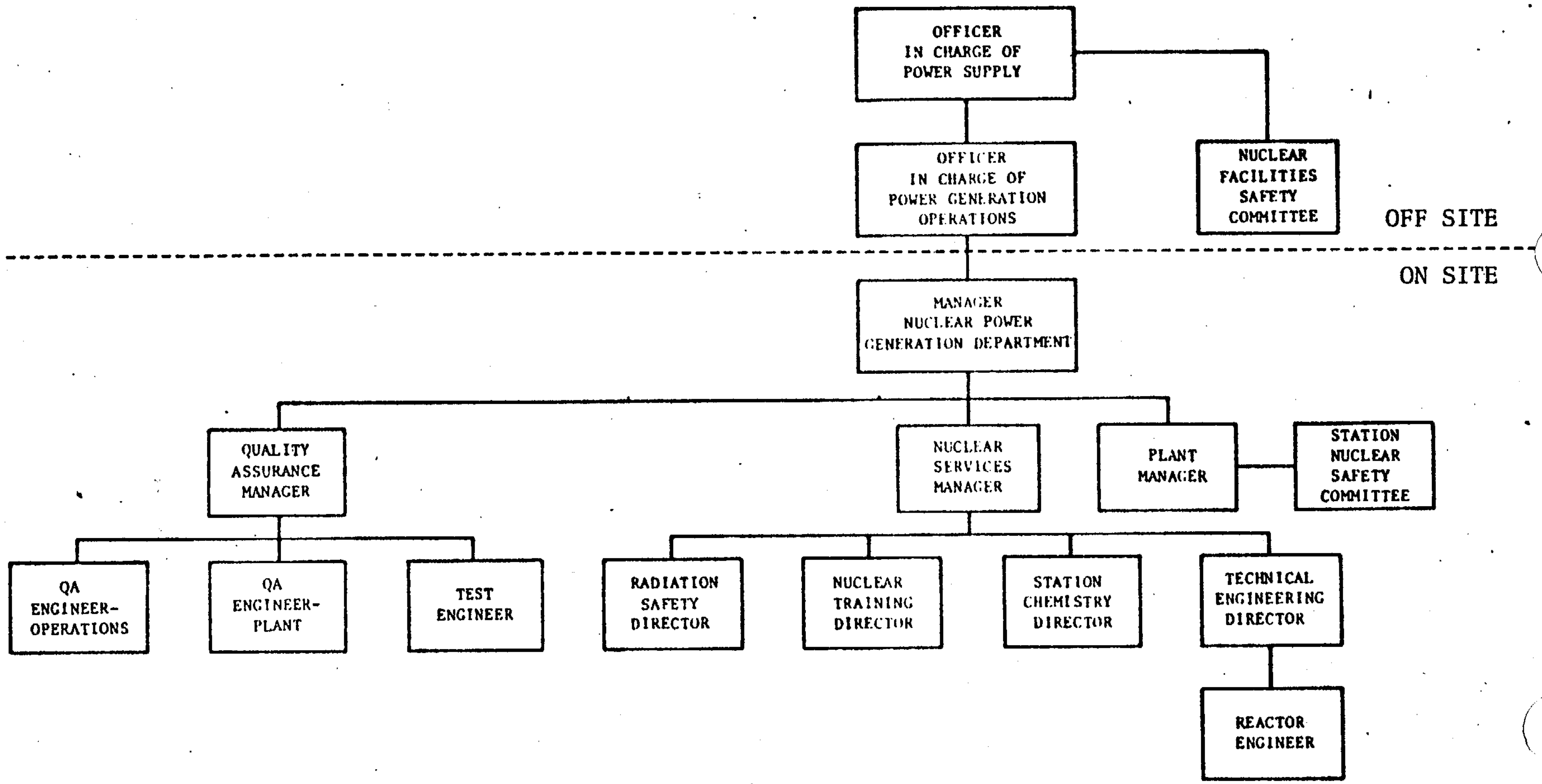

Figure 6.2-1 Facility Management and Technical Support Organization

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NFSC Chairman.

MEETING FREQUENCY

6.5.2.5 The NFSC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of NFSC shall consist of the Chairman or his designated alternate and a majority of the NFSC members including alternates. In the event both the Chairman and the Vice Chairman are absent, one of the permanent voting members will serve as Acting Chairman. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The following subjects shall be reported to and reviewed by the Committee insofar as they relate to matters of nuclear safety:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or expeiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

REVIEW (Continued)

g. Reportable Occurrences, as defined in Regulatory Guide 1.16, Revision 4.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meeting minutes of the Station Nuclear Safety Committee.

AUDITS

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the NFSC. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b. The performance, training and qualifications of the entire facility staff at least once per year.

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per two years.

e. The Facility Emergency Plan and implementing procedures at least once per two years.

f. The Facility Security Plan and implementing procedures at least once per two years.

g. Any other area of facility operation considered appropriate by the NFSC or the Senior Company Officer in charge of Power Supply.

AUTHORITY

6.5.2.9 The NFSC shall report to and advise the Senior Company Officer in charge of Power Supply on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

Amendment No. 16

RECORDS

6.5.2.10 Records of NFSC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NFSC meeting shall be prepared, approved and forwarded to the Senior Company Officer in charge of Power Supply within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 e, f, g and h above, shall be prepared, approved and forwarded to the Senior Company Officer in charge of Power Supply within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Company Officer in charge of Power Supply and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken in the event of a Reportable Occurrence.

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each Reportable Occurrence Report submitted to the Commission shall be reviewed by the SNSC and submitted to the NFSC Chairman, the Plant Manager and the Manager, Nuclear Power Generation Department.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with immediately.

b. The Safety Limit violation shall be reported to the Commission, the Manager, Nuclear Power Generation Department and to the NFSC Chairman immediately.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the NFSC Chairman and the Manager, Nuclear Power Generation Department within 10 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USAEC Regulatory Guide 1.33 (issued November 1972) except as provided in 6.8.2 and 6.8.3 below.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and any changes to them shall be reviewed and approved for implementation in accordance with a written administrative control procedure approved by the Manager, Nuclear Power Generation Department, with the concurrence of the Station Nuclear Safety Committee and the Nuclear Facilities Safety Committee. The administrative control procedure required by this specification shall, as a minimum, require that:

a. Each proposed procedure/procedure change involving safety related components and/or operation of same receives a pre-implementation review by the SNSC except in case of an emergency.

b. Each proposed procedure/procedure change which renders or may render the Final Safety Analysis Report or subsequent safety analysis reports inaccurate and those which involve or may involve potential unreviewed safety questions are approved by the SNSC prior to implementation.

c. The approval of the Nuclear Facilities Safety Committee shall be sought if, following its review, the Station Nuclear Safety Committee finds that the proposed procedure/procedure change either involves an unreviewed safety question or if it is in doubt as to whether or not an unreviewed safety question is involved.

6.8.3 A mechanism shall exist for making temporary changes and they shall only be made by approved management personnel in accordance with the requirements of ANSI 18.7-1972. The change shall be documented, and reviewed by the SNSC within 7 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE AND REPORTABLE OCCURRENCE REPORTS

6.9.1 Information to be reported to the Commission, in addition to the reports required by Title 10, Code of Federal Regulations, shall be in accordance with the Regulatory Position in Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications".

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of Region 1, Office of Inspection and Inforcement within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Each containment integrated leak rate test shall be the subject of a summary technical report including results of the local leak rate tests since the last report. The report shall include analyses and interpretations of the results which demonstrate compliance in meeting the leak rate limits specified in the Technical Specifications.

b. A report covering the X-Y xenon stability tests within three months upon completion of the tests.

c. To provide the Commission with added verifications of the safety and reliability of the pre-pressurized Zircaloy-clad nuclear fuel, a limited program of non-destructive fuel inspections will be conducted. The program shall consist of a visual inspection (e.g., underwater TV, periscope, or other) of the two lead burnup assemblies in each region during the first, second, and third refueling shutdowns. Any condition observed by this inspection which would lead to unacceptable fuel performance may be the object of an expanded surveillance effort. If another domestic plant which contains pre-pressurized fuel of a similar design reaches fuel exposures equal to or greater than at Indian Point Unit No. 2, and if a limited inspection program is or has been performed there, then the program may not have to be performed at Indian Point Unit No. 2. However, such action requires approval of the Nuclear Regulatory Commission. The results of these inspection will be reported to the Nuclear Regulatory Commission.

d. A written report shall be forwarded within 30 days to the Division of Reactor Licensing and to the Director of the Region 1, Office of Inspection and Enforcement, in the event of:

   1. Discovery of the release of radioactive liquids excluding tritium and dissolved noble gases exceeding 5 curies from the site during a consecutive 3 calendar month period.

   2. Discovery of the release of radioactive gases exceeding 50% of the limits specified in Specification 3.9.B.3.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety

c. Reportable Occurrence Reports.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all source material of record.

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

SAFETY EVALUATION BY THE OFFICE OF

NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 16 TO FACILITY LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY

OF NEW YORK, INCORPORATED

INDIAN POINT NUCLEAR GENERATING

UNIT NO. 2

DOCKET NO. 50-247

Introduction

By letter dated January 6, 1976, Consolidated Edison Company of New York, Inc. (Con Ed) proposed a change to the Technical Specifications appended to Facility Operating License No. DPR-26 for Indian Point Nuclear Generating License No. DPR-26 for Indian Point Nuclear Generating Unit No. 2. The proposed changes involve changes to the reporting requirements.

Discussion

The proposed changes would update the reporting requirements to be consistent with the present Regulatory guidance (Regulatory Guide 1.16, Revision 4). At the present time, reporting is accomplished in accordance with Regulatory Guide 1.16, Revision 3. The change, therefore, involves only the differences between Revision 3 and Revision 4 of Regulatory Guide 1.16. The difference between Revision 3 and Revision 4 of Regulatory Guide 1.16 consists of a change from the use of the term "abnormal occurrence" to the use of the term "reportable occurrence." Revision 4 also incorporates some minor changes in the distribution of reports from the licensees.

In Section 208 of the Energy Reorganization Act of 1974 "abnormal occurrence" is defined as an unscheduled incident or event which the Commission determines is significant from the standpoint of public health and safety. The term "abnormal occurrence" is reserved for usage by NRC. Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications," Revision 4, enumerates required reports consistent with Section 208 and identifies the reports required of all licensees not already identified by the regulations.

Evaluation

Con Ed proposes to change its Technical Specifications to be consistent with present Regulatory guidance on reporting. Their reporting will, therefore, be consistent with reports received from other licensees and will permit more rapid recognition of potential problems. No changes are proposed in the requirements associated with operation of the facility and; therefore, there is no increase in the probability or consequences of accidents and no decrease in any safety margin.

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 statement, negative declaration, or 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 change 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 change 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: JAN. 8 1976

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

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 16 to Facility Operating License No. DPR-26 issued to Consolidated Edison Company of New York, Inc. which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 2, located in Buchanan, Westchester County, New York. The amendment becomes effective 30 days after the date of issuance.

This amendment revises the reporting requirements of the Technical Specifications for the Indian Point Nuclear Generating Unit No. 2.

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 Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is 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 statement, negative declaration or 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 January 6, 1976, (2) Amendment No. 16 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 Hendrick Hudson Free Library, 31 Albany Post Road, Montrose, New York 10548.

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 Reactor Licensing.

Dated at Bethesda, Maryland, this 8th day of January 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Reactor Licensing