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

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Posted

September 18, 1979

Am-59 to DPR-26

Docket Nos. 50-3
and 50-247

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

Dear Mr. Cahill:

The Commission has issued the enclosed Amendment No. 27 to Provisional Operating License No. DPR-5 for the Indian Point Station Unit No. 1 and Amendment No. 59 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Station Unit No. 2. These amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated February 26, 1979.

These amendments revise the Technical Specifications concerning the organization for Unit Nos. 1 and 2, the reporting requirements for Unit No. 1, the qualifications of the Chemistry and Radiation Safety Director for Unit Nos. 1 and 2, and the number of fire detectors for Unit No. 2.

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

Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 27 to DPR-5
2. Amendment No. 59 to DPR-26
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

cc: White Plains Public Library
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-3

INDIAN POINT STATION UNIT NO. 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 27
License No. DPR-5

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 February 26, 1979, 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 3.B of Provisional Operating License No. DPR-5 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 27, 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 da e of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 18, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 27

PROVISIONAL OPERATING LICENSE NO. DPR-5

DOCKET NO. 50-3

Revise Appendix A as follows:

Remove Pages	Insert Pages
3 through 5	3 through 5
8 through 12	8 through 11

3.0 Administrative and Procedural Safeguards

3.1 Organization

3.1.1 The organization for facility management and technical support shall be as shown in Figure 3.1.

3.1.2 The Facility Organization shall be as shown in Figure 3.2. The Support Facilities Supervisor is responsible for operations at the Unit No. 1 facility.

3.1.3 The Department Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

3.1.4 The operation of the facility, the operating organization, the procedures for operation, and modifications to the facility shall be subject to review by the Station Nuclear Safety Committee. The committee shall report to the Department Manager.

3.1.5 The Nuclear Facilities Safety Committee shall function or provide independent review and audit of designated activities in areas of nuclear engineering, chemistry, radiochemistry, metallurgy and non-destructive testing, instrumentation and control, radiological safety, mechanical and electrical engineering, administrative controls and quality assurance practices, and radiological environmental effects.

3.1.6 All fuel handling shall be under the direct supervision of a licensed operator.*

3.2 Operating Instructions and Procedures

3.2.1 No fuel will be loaded into the reactor core or moved into the reactor containment building without prior review and authorization by the Nuclear Regulatory Commission.

3.2.2 Detailed written instructions setting forth procedures used in connection with the operation and maintenance of the nuclear power plant shall conform to the Technical Specifications.

3.2.3 Operation and maintenance of equipment related to safety when there is no fuel in the reactor shall be in accordance with written instructions.

3.2.6 Radiation control procedures shall be maintained and made available to all station personnel. Station operation shall adhere to these procedures. These procedures show permissible radiation exposure, and shall be consistent with the requirements of 10CFR20. The radiation protection program shall be organized to meet the requirements of 10CFR20.

*Licensed operator for IP-1 or IP-2.

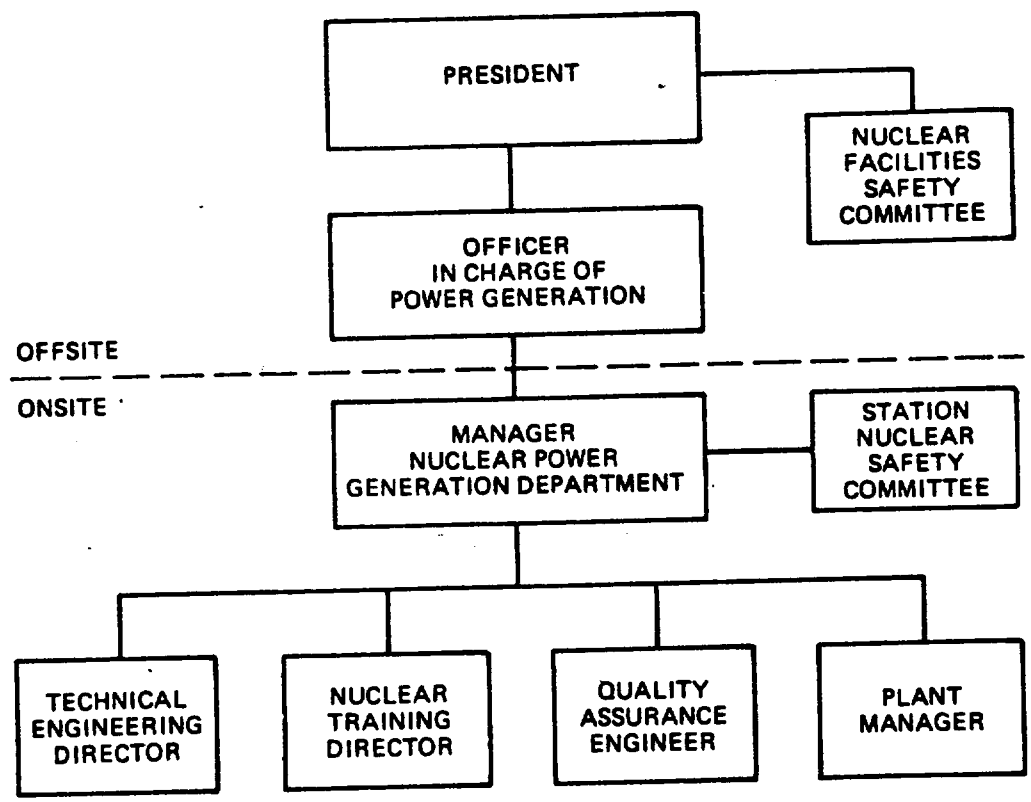

Figure 3.1 Facility Management and Technical Support Organization

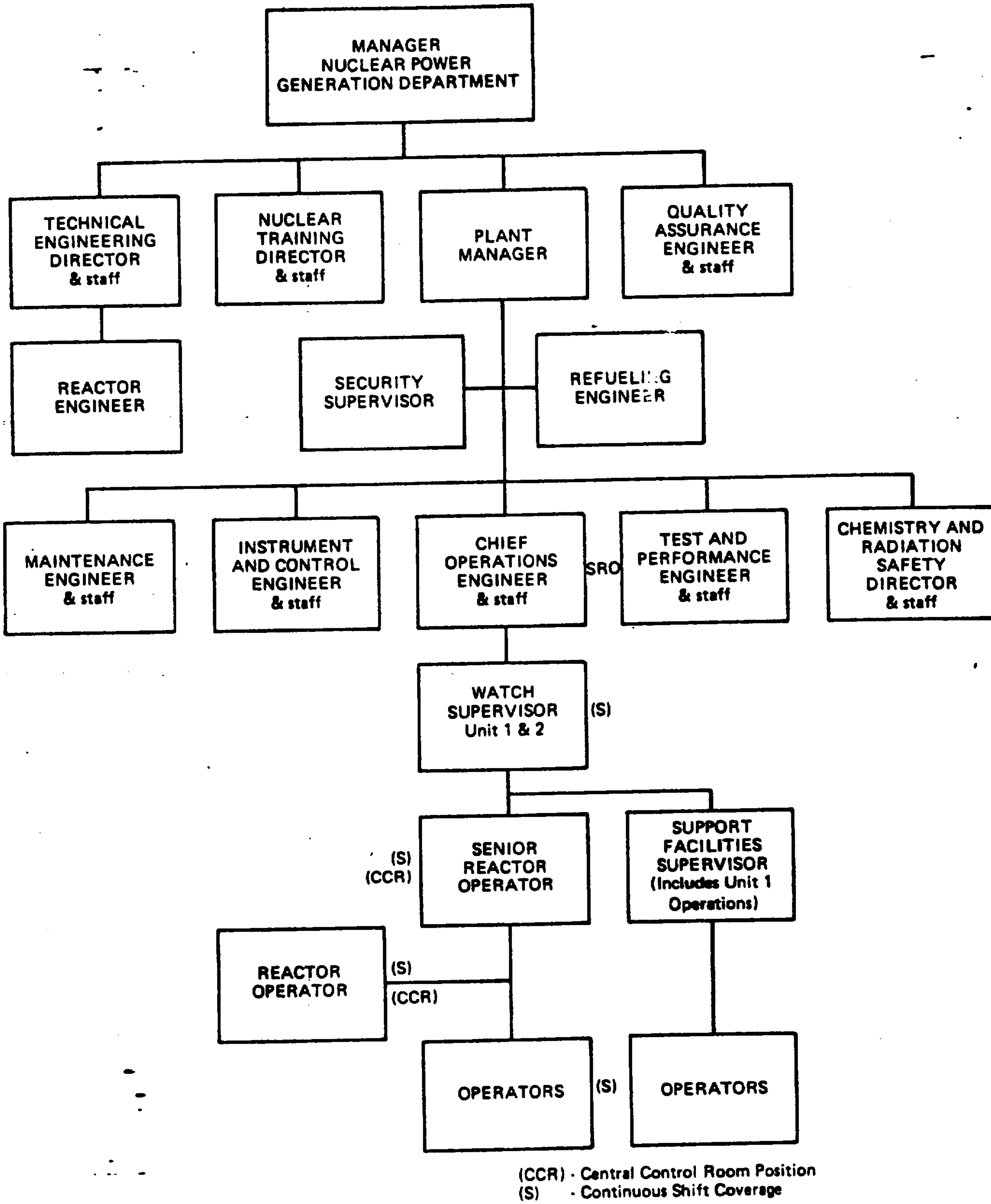

Figure 3.2 Facility Organization

6.5 The annual radiation exposure reports shall provide a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[2] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

Special Reports

6.6 Reports of major safety-related corrective maintenance shall be submitted to the Director, Office of Management Information and Program Control, with 40 copies to the Office of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, no later than 6 months following completion of such maintenance.

6.7 Each such report shall include a description of any major safety-related corrective maintenance performed including the system and component involved.

Reportable Occurrences

6.8 The reportable occurrences of specifications 6.8.1 and 6.8.2 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

Prompt Notification with Written Followup Report

6.8.1 The types of events listed below shall be reported within 24 hours of identification by telephone and confirmed by telegraph, mailgram, or facsimile

[2]This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

transmission to the Director of the Region I Office of Inspection and Enforcement or his designate, no later than the first working day following the event, with a written followup report within two weeks. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.[3]

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

[3] Leakage of packing, gaskets, mechanical joints or seal welds within the limits for identified leakage set forth in technical specifications need not be reported under this item.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

g. Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

Thirty Day Written Reports

6.8.2 The types of events listed below shall be the subject of written reports to the Director of the Region I Office of Inspection and Enforcement within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.[4]

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.[4]

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.8.1.c above designed to contain radioactive material resulting from the fission process.[5]

6.9 Any references to the term "Safety Analysis Report", "SAR" or "FSAR" for Indian Point Station, Unit No. 1, shall be deemed to refer, as appropriate, to the following exhibits which are a part of the application: F-4 (Rev.-3), F-6 (Rev.-2), F-7 (Rev.-1), G-3 (Rev.-2), H-14 (Rev.-2), K-4, K-4A, K-4B, K-5 (Rev.-1, but not including Sections 2.1.2 through 2.3.7.4, Section 4, Figures 2-1 through 2-9, Figure 3-17, Figures 4-1 through 4-12, and Appendix A), K-5A1, K-5A10, K-5A11, K-5A11A, K-5A12, K-5A13, K-5A14, as amended, K-5A15, K-16, and the document entitled "Final Hazards Summary Report for the Consolidated Edison Indian Point Reactor Core B", as amended.

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[4]Routine surveillance testing, instrument calibration, or preventive maintenance which require system configurations as described need not be reported except where test results themselves reveal a degraded mode as described.

[5]Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

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. 59
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 February 26, 1979, 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. 59 , 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 18, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 59

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.1-18	3.1-18
Table 3.13-1	Table 3.13-1
Table 4.1-1	Table 4.1-1
6-1	6.1
6-2	6-2
6-3	6-3
6-5	6-5
6-6	6-6
6-7	6-7
6-11	6-11

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 Department Manager or his designated alternate. Under these conditions, an allowable primary system leakage rate of 10 gpm 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

Table 3.13-1

Fire Detection Instruments

Instrument Location	Minimum Instruments Operable Heat	Minimum Instruments Operable Smoke (ionization detectors)
1. Central Control Room (Control Building: El-53')	N/A	4
2. Cable Spreading Room (Control Building: El-33')	N/A	7
3. Switchgear Room (Control Building: El-15')	N/A	7
4. Electrical Tunnel (El-33' to El-68')	38*	3
5. Electrical and Piping Tunnel and Piping Penetration Area (PAB and Fan House: El-68' to El-51')	N/A	2
6. Electrical Penetration Area (Fan House: El-46')	N/A	4
7. Diesel Generator Building (El-67')	11	N/A
8. Boric Acid Transfer Pump Area (PAB: El-80')	N/A	1
9. Containment Spray Pump Area (PAB: El-68')	N/A	3
10. Containment Fan Cooler Units (Containment: El-68')	4 per FC unit	N/A

*temperature detector/trip devices

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTS OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
1.	Nuclear Power Range	S	D (1) M* (3)	M (2)	1) Heat balance calibration 2) Signal to ΔT; bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial off-set
2.	Nuclear Intermediate Range	S (1)	N.A.	P**(2)	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trip)
3.	Nuclear Source Range	S (1)	N.A.	P**(2)	1) Once/shift when in service 2) Bistable action (alarm, trip)
4.	Reactor Coolant Temperature	S	R	M (1) (2)	1) Overtemperature-ΔT 2) Overpower-ΔT
5.	Reactor Coolant Flow	S	R	M	
6.	Pressurizer Water Level	S	R	M	
7.	Pressurizer Pressure (High and Low)	S	R	M	
8.	6.9 Kv Voltage & Frequency	N.A.	R	M	Reactor protection circuits only
9.	Analog Rod Position	S	R	M	

* By means of the moveable incore detector system. The January 1976 scheduled measurements with the moveable incore detector system may be delayed until February 6, 1976.

** Prior to each reactor startup if not done previous week.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Department Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

FACILITY MANAGEMENT AND TECHNICAL SUPPORT

6.2.1 The organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown, and during recovery from reactor trips.

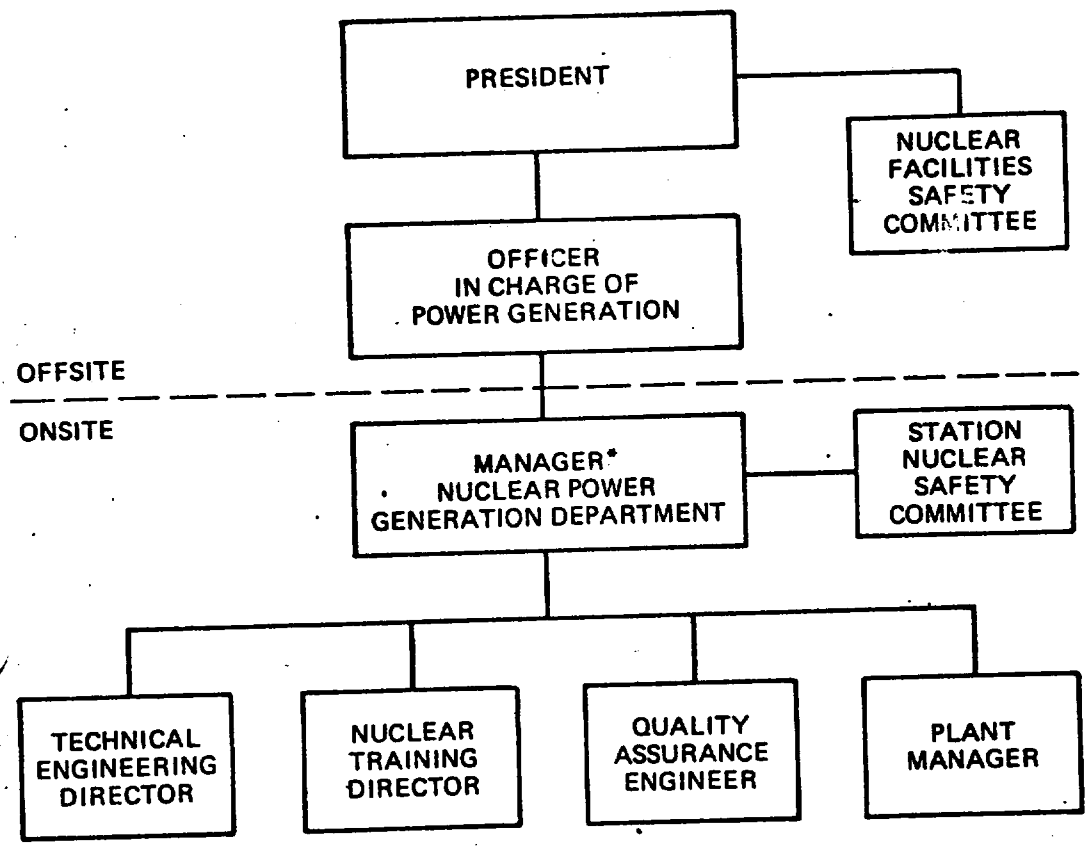
d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. ALL CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling. This individual shall have no other concurrent responsibilities during this operation.

f. A Fire Brigade of at least five members shall be maintained on site at all times. This excludes four members of the minimum shift crew necessary for safe shutdown of the plant and any personnel required for other essential functions during a fire emergency. During periods of cold shutdown, the Fire Brigade will exclude two members of the minimum shift crew.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Chemistry and Radiation Safety Director who shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975.



*Responsible for performance and monitoring of the Fire Protection Program.

Figure 6.2-1 Facility Management and Technical Support Organization

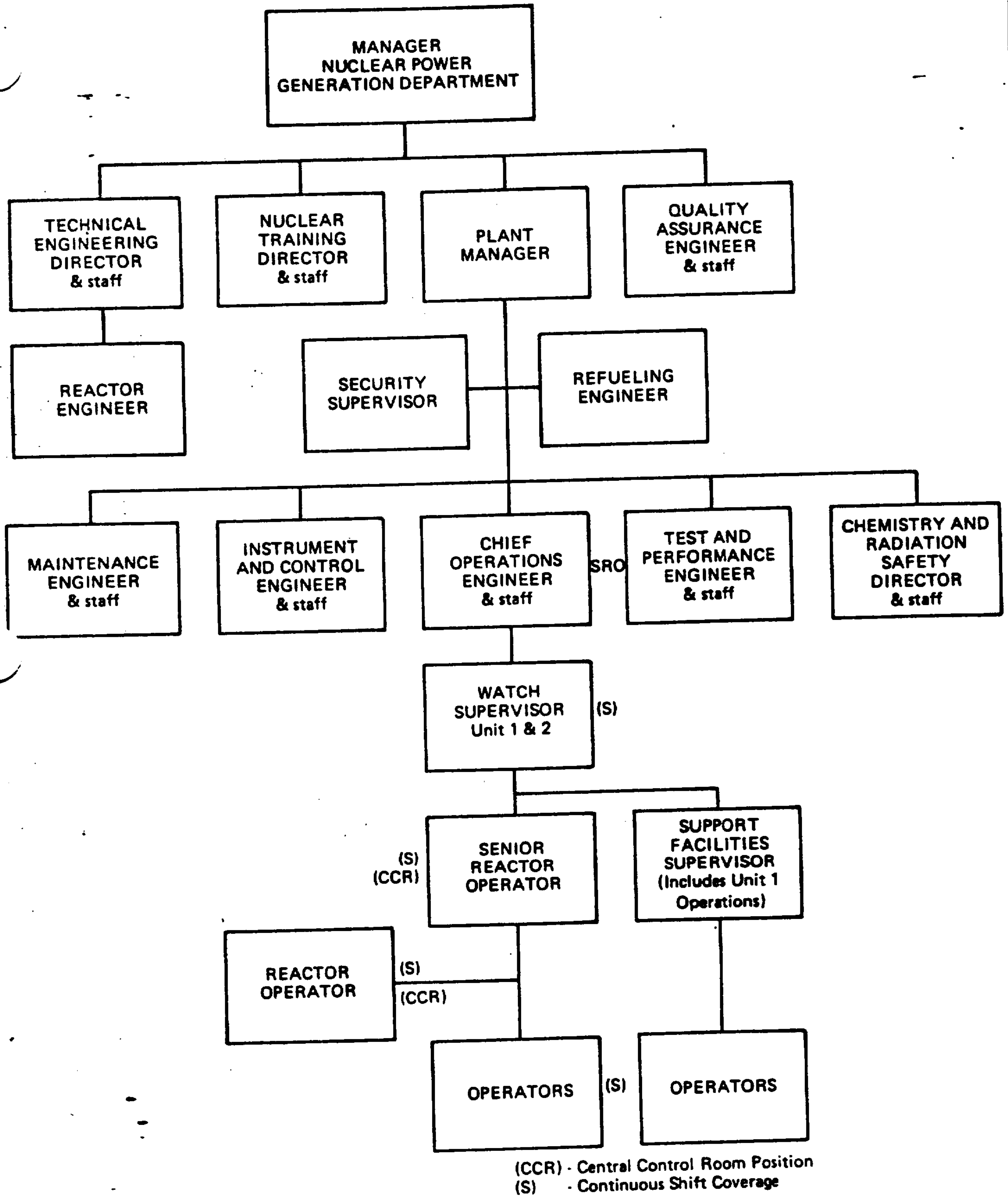


Figure 6.2-2 Facility Organization

6.3.2 The Department Manager shall meet or exceed the minimum qualifications specified for Plant Manager in ANSI N18.1-1971.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Training Director and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Nuclear Training Director and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976 with the exception of the training program schedule.

6.5 REVIEW AND AUDIT

6.5.1 STATION NUCLEAR SAFETY COMMITTEE (SNSC)

FUNCTION

6.5.1.1 The Station Nuclear Safety Committee shall function to advise the Department Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Station Nuclear Safety Committee shall be composed as follows:

Chairman:	Plant Manager
Member:	Technical Engineering Director
Member:	Quality Assurance Engineer
Member:	Chief Operations Engineer
Member:	Security Supervisor
Member:	Test and Performance Engineer
Member:	Instrument and Control Engineer
Member:	Maintenance Engineer
Member:	Chemistry and Radiation Safety Director
Member:	Reactor Engineer
Member:	Department Manager
Member:	Refueling Engineer

ALTERNATES

6.5.1.3 Alternate members shall be appointed in writing by the SNSC Chairman to serve on a temporary basis.

MEETING FREQUENCY

6.5.1.4 The SNSC shall meet at least once per calendar month and as convened by the SNSC Chairman.

QUORUM

6.5.1.5 A quorum of the SNSC shall consist of the Chairman or Vice Chairman and five members including no more than two alternates.

RESPONSIBILITIES

6.5.1.6 The Station Nuclear Safety Committee shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, and 2) any other proposed procedures or changes thereto as determined by the Department Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications and preparation and forwarding of a report covering evaluation and recommendations to prevent recurrence to the Manager, Nuclear Power Generation Department and to the Chairman of the Nuclear Facilities Safety Committee.

f. Review of facility operations to detect potential safety hazards.

g. Performance of special reviews and investigations and the issuance of reports thereon as required by the Chairman of the Nuclear Facilities Safety Committee.

h. Review of the Plant Security Plan and implementing procedures and submission of recommended changes to the Chairman of the Nuclear Facilities Safety Committee.

i. Review of the Emergency Plan and implementing procedures and submission of recommended changes to the Chairman of the Nuclear Facilities Safety Committee.

AUTHORITY

6.5.1.7 The Station Nuclear Safety Committee shall:

a. Recommend to the Department Manager, in writing, approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

AUTHORITY (Continued)

c. Provide immediate written notification to the Chairman, Nuclear Facilities Safety Committee of disagreement between the recommendations of the SNSC and the actions contemplated by the Department Manager. However, the course of action determined by the Department Manager pursuant to 6.1.1 above shall be followed.

RECORDS

6.5.1.8 The Station Nuclear Safety Committee shall maintain written minutes of each meeting and copies shall be provided to, as a minimum, the Manager, Nuclear Power Generation Department and the Chairman, Nuclear Facilities Safety Committee.

6.5.2 NUCLEAR FACILITIES SAFETY COMMITTEE (NFSC)

FUNCTION

6.5.2.1 The Nuclear Facilities Safety Committee shall function to provide independent review and audit of designated activities in the areas of:

a. reactor operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy and non-destructive testing

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. administrative controls and quality assurance practices

i. radiological environmental effects

j. other appropriate fields associated with the unique characteristics of the nuclear power plant

h. The Facility Fire Protection Program and implementing procedures at least once per 24 months.

i. A fire protection and loss prevention inspection and audit shall be performed utilizing either qualified offsite licensee personnel or an outside fire protection firm at least once per 12 months.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at least once per 36 months.

k. The environmental surveillance program pertaining to radiological matters and implementing procedures at least once per 12 months.

l. Any other area of facility operation considered appropriate by the NFSC or the President of the Company.

AUTHORITY

6.5.2.9 The NFSC shall report to and advise the President of the Company on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

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 President and to Senior Company Officers concerned with nuclear facilities 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 President and to Senior Company Officers concerned with nuclear facilities 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 Officers concerned with nuclear facilities and to the management positions responsible for the areas audited within 30 days after completion of the audit.

TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

Monthly Operating Report

6.9.1.5 Routine reports of operating statistics, operating and shutdown experience and major safety-related corrective maintenance shall be submitted on a monthly basis to the Director, Office of Management Information & Program Control, with 40 copies to the Office of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, no later than 15 days following the calendar month covered by the report.

6.9.1.6 Each monthly operating report shall include:

a. A tabulation of plant operating data and statistics.

b. A narrative summary of operating experience during the report period relating to safe operation of the facility, including major safety-related corrective maintenance not covered in 6.9.1.6.c.5 below.[3]

c. For each outage or forced reduction in power[4] of over twenty percent of rated power where the reduction extends for greater than four hours:

1. The proximate cause and the system and major component involved (if the outage or forced reduction in power involved equipment malfunction);
2. A brief discussion of (or reference to reports of) any reportable occurrences pertaining to the outage or power reduction;
3. Corrective action taken to reduce the probability of recurrence, if appropriate;
4. Operating time lost as a result of the outage or power reduction (for scheduled or forced outages,[5] use the generator off-line

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[3]Any safety-related maintenance information not available for inclusion in the monthly operating report for a report period shall be included in a subsequent monthly operating report not later than 6 months following completion of such maintenance.

[4]The term "forced reduction in power" is defined as the occurrence of a component failure or other condition which requires that the load on the unit be reduced for corrective action immediately or up to and including the very next weekend. Note that routine preventive maintenance, surveillance and calibration activities requiring power reductions are not covered by this section.

[5]The term "forced outage" is defined as the occurrence of a component failure or other condition which requires that the unit be removed from service for corrective action immediately or up to and including the very next weekend.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 27 TO PROVISIONAL OPERATING LICENSE NO. DPR-5
AND AMENDMENT NO. 59 TO FACILITY OPERATING LICENSE NO. DPR-26
CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT STATION UNIT NOS. 1 AND 2
DOCKET NOS. 50-3 AND 50-247

Introduction

By letter dated February 26, 1979, Consolidated Edison Company of New York, Inc. (the licensee) requested amendments to Facility License Nos. DPR-5 and DPR-26 for the Indian Point Unit Nos. 1 and 2, respectively. The proposed changes shift some of the Plant Manager's responsibilities to the Department Manager, require that the Nuclear Facilities Safety Committee report to the President of the Company, and replace the Monthly Operating Report required for Unit No. 1 with a special report. In addition, the qualifications of the Chemistry and Radiation Safety Director have been added after discussions with the licensee.

Evaluation

The proposed changes to Section 3.1 for Unit No. 1 and Sections 3.1, 6.1, 6.2 and 6.5 for Unit No. 2 would accomplish the following: (1) shift the responsibility for overall facility operation from the Plant Manager to the Department Manager; (2) require that the Station Nuclear Safety Committee report to the Department Manager rather than the Plant Manager; and (3) make the Nuclear Facilities Safety Committee responsible to the President of the Company rather than the Senior Company Officer in Charge of Power Supply. In all these cases, the change shifts responsibility to a higher level of management which we find acceptable.

The organization charts (Figures 3.1 and 3.2 for the Unit No. 1 and Figures 6.2-1 and 6.2-2 for Unit No. 2) have been changed to reflect the above changes, as well as to delete reference to Unit No. 3 which is no longer owned or operated by the licensee.

The proposed changes to Sections 6.6, 6.7 and 6.8 for Unit No. 1 replace the requirement for the Monthly Operating Report with a special report. The special report proposed retains the requirement to describe any major safety-related corrective maintenance, but deletes the other information. We find this acceptable because this deleted information is meaningless as long as Unit No. 1 is shut down and defueled. However, the requirement for the Monthly Operation Report will have to be reinserted should Unit No. 1 be started up again.

The qualifications of the Chemistry and Radiation Safety Director have been added to Section 6.3 of the Unit No. 2 Technical Specifications. Reference to Regulatory Guide 1.8, September 1975, satisfies our position and is acceptable. (This section is now identical to the Standard Technical Specifications for all light water reactors.)

Table 4.1-1 was modified to reinsert a footnote that was inadvertently omitted in issuing Amendment No. 51 dated March 19, 1979.

The licensee, by letter dated August 3, 1979 to Mr. Boyce Grier, stated that the original number of heat detectors associated with the Containment Fan Cooler Units was incorrect. Table 3.13-1 has been revised to reflect this error. We find the number of heat detectors now specified to be adequate to detect any overheating condition. Therefore, the proposed change is acceptable.

Environmental Consideration

We have determined that the amendments do 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 amendments involve 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 these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: September 18, 1979

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-3 AND 50-247

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

NOTICE OF ISSUANCE OF AMENDMENT TO OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 27 to Provisional Operating License No. DPR-5, and Amendment No. 59 to Facility Operating License No. DPR-26 issued to the Consolidated Edison Company of New York (the licensee), which revised Technical Specifications for operation of the Indian Point Station Unit No. 1 and Indian Point Nuclear Generating Unit No. 2 (the facilities) located in Buchanan, Westchester County, New York. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications concerning the operating organization for Unit Nos. 1 and 2, the reporting requirements for Unit No. 1, the qualifications of the Chemistry and Radiation Safety Director for Unit Nos. 1 and 2, and number of fire detectors for Unit No. 2.

The application for the amendments 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 amendments. Prior public notice of the amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments 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 these amendments.

For further details with respect to this action, see (1) the application for amendments dated February 26, 1979, (2) Amendment Nos. 27 and 59 to License Nos. DPR-5 and DPR-26, respectively, 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 Operating Reactors.

Dated at Bethesda, Maryland, this 18th day of September, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors