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April 22, 1985

Posted
Amdt. 94
to DPR-26

Docket Nos. 50-003
and 50-247

Distribution
~~NRC PDR~~
ORB#1 RDG
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LHarmon
BGrimes
TBarnhart (8)
DBrinkman
OPA, CMiles
ORB#1 Gray file (4)
Docket file
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MSlosson
OELD
EJordan
JPartlow
WJones
ACRS (10)
RDiggs

Mr. John D. O'Toole
Vice President
Nuclear Engineering 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. 34 to Facility Operating License No. DPR-5 and Amendment No. 94 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Station Unit Nos. 1 and 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated June 20, 1984.

The amendment revises the Indian Point Units No. 1 and No. 2 Technical Specifications to incorporate the reporting requirements in sections 50.72 and 50.73 of Title 10 Code of Federal Regulations. Section 50.72 revises the immediate notification requirements for operating nuclear power plants. Section 50.73 provides for a revised Licensee Event Report

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

/s/JDNeighbors

Joseph D. Neighbors, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 34 to DPR-5
2. Amendment No. 94 to DPR-26
3. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:DL	ORB#1:DL	ORB#1:DL	C-ORB#1:DL	OELD	AD:OR:DL
CParrish	MSlosson/ts	DNeighbors	SVarga		GLainas
04/1/85	04/3/85	04/3/85	04/ /85	04/10/85	04/ /85

4/10

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

CONSOLIDATED EDISION COMPANY OF NEW YORK, INC.

DOCKET NO. 50-3

INDIAN POINT NUCLEAR GENERATING UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 34
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 June 20, 1984, 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-5 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 34 , 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: April 22, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 34 TO FACILITY OPERATING LICENSE NO. DPR-5

DOCKET NO. 50-3

Revise Appendix A as follows:

Remove Pages	Insert Page
12	12
13	--
14	--

Special Reports

6.10 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.

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

REPORTABLE EVENT ACTION

6.12.0 A Reportable Event is defined as any of the conditions specified in 10 CFR 50.73.a(2)

6.12.1 The following actions shall be taken in the event of a Reportable Event:

a. A report shall be submitted to the Commission pursuant to the requirements of 10 CFR 50.73 and

b. Each Reportable Event report submitted to the Commission shall be submitted to the NFSC Chairman, and the Vice President-Nuclear Power and be reviewed by the SNSC.

6.13 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.

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

CONSOLIDATED EDISION COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94
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 June 20, 1984, 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. 94 , 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

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 22, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Page
ii	ii
iv	iv
3.10-8	3.10-8
3.13-1	3.13-1
3.13-2	3.13-2
3.13-3	3.13-3
3.13-4	3.13-4
------	3.13-5
3.14-1	3.14-1
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-----	4.9-2
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be logged once per shift and after a load change greater than 10 percent of rated power.

Basis

Design criteria have been chosen for normal operations, operational transients and those events analyzed in FSAR Section 14.1 which are consistent with the fuel integrity analyses. These related to fission gas release, pellet temperature and cladding mechanical properties. Also the minimum DNBR in the core must be less than 1.30 in normal operation or in short term transients.

In addition to the above conditions, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant accident analysis based on the ECCS acceptance criteria limit of 2200°F. This is required to meet the initial conditions assumed for loss of coolant accident. To aid in specifying the limits on power distribution the following hot channel-factors are defined.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

3.13 FIRE PROTECTION AND DETECTION SYSTEMS

Applicability

This specification applies to the operability of fire protection and detection systems provided for protection of safe shutdown systems.

Objective

To assure the operability of fire protection and detection systems.

Specification

A. High Pressure Water Fire Protection System

1. The high pressure water fire protection system shall have:

   a. Two (2) main motor-driven fire pumps and one diesel-driven fire pump operable and properly aligned to the high pressure fire header.

   b. A minimum available water volume 360,000 gallons contained in the City Water Tank and 300,000 gallons contained in the Fire Water Tank for fire protection purposes.

   c. All piping and valves necessary for proper functioning of any portion of the system required for protection of safe shutdown systems operable.

2. The requirements of specification 3.13.A.1 may be modified to allow any one of the following conditions to exist at any one time. If the inoperable equipment is not restored to operable status within the time period specified then, in lieu of any other report required by 10 CFR 50.73 a Special Report shall be prepared and submitted to the Commission pursuant to specification 6.9.2.b within the next thirty (30) days outlining the plans and procedures to be used for restoring the inoperable equipment to operable status or for providing an alternate pumping capability or water supply.

   a. One or both motor-driven fire pumps and/or one water supply may be out of service provided the inoperable equipment is restored to operable status within seven (7) days.

   b. The diesel-driven fire pump and/or one water supply may be out of service provided the inoperable equipment is restored to operable status within (7) days.

3. With the high pressure water fire protection system inoperable in a manner other than permitted by specification 3.13.A.2:

   a. An alternate fire protection system shall be established within 24 hours.

b. In lieu of any other report required by 10 CFR 50.73

(i) The NRC Region I Office shall be notified within 24 hours of identification by telephone and confirm by telegraph, mailgram or facsimile transmission no later than the first working day following the event; and

(ii) A Special Report shall be submitted in accordance with specification 6.9.2.b within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

c. If the requirement of 3.13.A.3.a cannot be satisfied within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirement of 3.13.A.3.a cannot be satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

B. Fire Protection Spray Systems

1. The following spray systems shall be operable whenever equipment in the area is required to be operable:

a. Electrical Tunnel Fire Protection Water Spray System (El-33' in Control Building to El-68' in PAB).

b. Diesel Generator Building Water Spray System (El-67' in D.G. Building).

c. Containment Fan Cooler Charcoal Filter Dousing System (El-68' in Containment).

2. If the requirements of 3.13.B.1 cannot be satisfied:

a. Additional equivalent capacity fire hose(s) shall be routed to the affected area(s) from an operable hose station or hydrant within one (1) hour.

b. The inoperable equipment shall be restored to operable status within 14 days or, in lieu of any other report required by 10 CFR 50.73 a Special Report shall be prepared and submitted to the Commission pursuant to specification 6.9.2.b within the next 30 days outlining the cause of the inoperability and the plans for restoring the equipment to operable status.

C. Penetration Fire Barriers

1. The following penetration fire barriers shall be functional at all times:

   a. Penetration fire barriers between the central control floor and the cable spreading room.

   b. Penetration fire barriers between the 480 V switchgear room and the cable spreading room.

   c. Penetration fire barrier between the PAB and the electrical tunnel.

2. If the requirements of 3.13.C.1 cannot be satisfied:

   a. Within one (1) hour, either a continuous fire watch shall be established on at least one side of the affected penetration(s), or the operability of fire detectors on at least one side of the non-functional fire barrier(s) shall be verified and an hourly fire watch patrol shall be established.

   b. The non-functional penetration fire barrier(s) shall be restored to functional status within seven (7) days or, in lieu of any other report required by 10 CFR 50.73 a Special Report shall be prepared and submitted to the Commission pursuant to specification 6.9.2.b within the next 30 days outlining the cause of the malfunction and the plans for restoring the barrier(s) to functional status.

D. Fire Detection Systems

1. As a minimum, the fire detection instrumentation for each location shown in Table 3.13-1 shall be operable whenever equipment in that location is reuqired to be operable.

2. With the number of operable fire detection instruments less than the minimum required by Table 3.13-1:

   a. For instruments outside containment-a fire watch patrol shall be established within 1 hour to inspect the affected location(s) at a frequency of at least once per hour.

   b. For instruments inside containment-Either a fire watch patrol shall be established to inspect the affected location(s) at least once per eight (8) hours, or the containment air temperature shall be monitored at least once per hour.

   c. The minimum operable instrumentation required in Table 3.13-1 shall be restored within 14 days or, in lieu of any other report required by 10 CFR 50.73 a Special Report shall be prepared and submitted to the Commission pursuant to specification 6.9.2.b within the next 30 days outlining the cause of the malfunction and the plans for restoring the instrumentation to operable status.

E. Fire Hose Stations and Hydrants

1. The fire hose stations and fire hydrants shown in Table 3.13-2 shall be operable whenever safety-related equipment in the areas protected by the hose stations and hydrants is required to be operable.

2. If the requirements of 3.13.E.1 cannot be satisfied:

   a. Additional equivalent capacity fire hose(s) shall be routed to the affected area(s) from an operable hose station or hydrant within one (1) hour.

   b. The inoperable spray system(s) shall be restored to operable status within 14 days or, in lieu of any other report required by 10 CFR 50.73 a Special Report shall be prepared and submitted to the Commission pursuant to specification 6.9.2.b within the next 30 days outlining the cause of inoperability and the plans for restoring the spray system(s) to operable status.

F. Cable Spreading Room Halon System

1. The Cable Spreading Room Halon System shall be operable at all times with the halon storage tanks having an equivalent of at least 95% of full charge weight and an equivalent of at least 90% of full charge pressure at standard temperature and pressure (STP) conditions.

2. If the requirements of 3.13.F.1 cannot be satisfied:

   a. A continuous fire watch with backup fire protection equipment shall be established within 1 hour for the Cable Spreading Room.

   b. The Cable Spreading Room Halon System shall be restored to operable status within 14 days or, in lieu of any other report required by 10 CFR 50.73 a Special Report shall be prepared and submitted to the Commisson pursuant to specification 6.9.2.b within the next 30 days outlining the cause of the inoperability and the plans for restoring the Halon System to operable status.

Basis

These specifications are established to assure the operability of fire protection and detection systems provided to protect equipment utilized for safe shutdown of the unit. The fire protection and detection systems are described in Revision 1 to "Review of the Indian Point Station Fire Protection Program," submitted to the NRC by letter dated April 15, 1977, and also in the Fire Protection Safety Evaluation Report issued by the NRC Regulatory Staff in conjunction with Amendment No. 46 to DPR-26 on January 31, 1979.

3.14 HURRICANE ALERT

Applicability

Applies to a hurricane with winds in excess of 87 knots, when a Hurricane Warning has been issued for any coastal area south of Indian Point or east of Indian Point as far east as New Haven, Connecticut.

Objective

To define actions permitted after receipt of Hurricane Warnings.

Specifications

3.14.a: If the National Weather Service issues a Hurricane Warning for a hurricane with wind in excess of 87 knots (approximately 100 mph) within 500 nautical miles of the facility, a prompt report shall be made to the NRC Incident Response Center within 1 hour of receipt of that Hurricane Warning. This notification is in lieu of the reporting requirements of 10 CFR 50.73.

3.14b: If the National Weather Service issues a Hurricane Warning for a hurricane with winds in excess of 87 knots within 320 nautical miles of the facility and a Hurricane Warning is in effect for any coastal area south of Indian Point or any coastal area east of Indian Point as far east as New Haven, Connecticut; the hurricane direction, translational velocity and average wind speed shall be monitored at least every hour and the Unit shall be placed in the Hot Shutdown condition within (4) hours. Appropriate action shall be taken to ensure that the plant is in the Cold Shutdown condition prior to arrival on site of a hurricane with winds in excess of 87 knots.

4.9 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of reactivity anomalies within the reactor.

Specification

Following a normalization of the computed boron concentration as a function of burn-up, the actual boron concentration of the coolant shall be periodically compared with the predicted value.

Basis

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, with the control rod groups in the desire positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity is compared with that predicted. This process of normalization shall be completed early in core life. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be throughly investigated and evaluated. The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive rod in the fully withdrawn positon is always maintained.

B. Sealed sources are exempt from Specification 4.15.A when the source contains 100 microcuries or less of beta and/or gamma emitting material or 5 microcuries or less of alpha emitting material.

C. The leakage test shall be capable of detecting the presence of 0.005 microcuries of radioactive material on the test sample. If the test reveals the presence of 0.005 microcuries or more of removable contamination, the sealed source shall immediately be withdrawn from use and either decontaminated and repaired, or be disposed of in accordance with Commission regulations.

D. A Special Report shall be prepared and submittted to the Commission pursuant to Specification 6.9.2.c within 30 days if source leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

Basis

The objective of this specification is to assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed the allowable limits specified in the Code of Federal Regulations.

REVIEW (Continued)

g. Reportable Events, as specified by 10 CFR 50.73.

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.

j. Environmental surveillance program pertaining to radiological matters.

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 facilty operation to all provisions contained within the Radiological Technical Specifications (Appendix A) and applicable license conditions at least once per 12 months.

b. The conformance to all provisions contained within the Environmental Technical Specifications (Appendix B) pertaining to radiological matters and applicable license conditions at least once per 12 months.

c. The performance, training and qualifications of the entire facility staff at least once per 12 months.

d. 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 6 months.

e. 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 24 months.

f. The Facility Emergency Plan and implementing procedures at least once per 24 months.

g. The Facility Security Plan and implementing procedures at least once per 24 months.

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

6.6 REPORTABLE EVENT ACTION

6.6.0 A Reportable Event is defined as any of the conditions specified in 10 CFR 50.73a(2)

6.6.1 The following actions shall be taken in the event of a Report Event:

a. A report shall be submitted to the Commission pursuant to the requirements of to 10 CFR 50.73 and

b. Each Reportable Event Report submitted to the Commission shall be submitted to the NFSC Chairman, and the Vice President-Nuclear Power and be reviewed by the SNSC.

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 Vice President-Nuclear Power 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 Vice President-Nuclear Power within 10 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained covering the activities referenced below:

a. 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.

b. Process Control Program implementation.

c. Offsite Dose Calculation Manual implementation.

d. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 1.21, Revision 1, April 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

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 appropriate General Manager, 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 Final Safety Analysis Report or subsequent safety analysis report 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 Reports

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC Regional Administrator of the Region I Office unless otherwise noted.

4. Operating time lost as a result of the outage or power reduction (for scheduled or forced outages,[9] use the generator off-line hours; for forced reductions in power, use the approximate duration of operation at reduced power);

5. A description of major safety-related corrective maintenance performed during the outage or power reduction, including the systems and component involved and identifification of the critical path activity dictating the length of the outage or power reduction; and

6. A report of any single release of radioactivity or radiation exposure specifically associated with the outage which accounts for more than 10% of the allowable annual values.

[9]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.

Special Reports

6.9.2 Special reports shall be submitted to the NRC Regional Administrator of the Region I Office 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 test 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. Inoperable fire protection and detection equipment (Specification 3.13).

c. Sealed source leakage in excess of limits (Specification 4.15).

d. The complete results of the steam generator tube inservice inspection (Specification 4.13.C).

e. Radioactive effluents (Specification 3.9).

f. Radiological environment monitoring (Specification 4.11)

g. Meteorological monitoring instrumentation (Specification 3.15).

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 Event 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 or record.

6.10.2 The following record shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material releases to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

Record Retention (continued)

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the SNSC and the NFSC.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Record of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.11 Radiation Protection Program

Procedure for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 High Radiation Area

6.12.1 As an acceptable alternate to the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.12.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry to such areas and the keys shall be maintained under the administrative control of the Watch Supervisor on duty.

6.13 <u>Environmental Qualification</u>

6.13.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of License No. DPR-26 dated October 24, 1980.

6.13.2. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 <u>Process Control Program (PCP)</u>

6.14.1 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information:

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the (SNSC).

2. Shall become effective upon review and acceptance by the (SNSC).

6.15 <u>Offsite Dose Calculation Manual (ODCM)</u>

6.15.1 The ODCM shall be approved by the Commission prior to implementation.

6.15.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluation justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact the change has been revised and found acceptable by the (SNSC).

2. Shall become effective upon review and acceptance by the (SNSC).

6.16 <u>Major Changes to Radioactive Liquid, Gaseous and Solid Waste</u>

6.16.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change was made. The discussion of each change shall contain:

a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59.

b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

d. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change, which shows the expected maximum exposures to individual in the Unrestricted Area and to the general population that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the (SNSC).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 34 TO PROVISIONAL OPERATING LICENSE NO. DPR-5 AND

AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNITS 1 AND 2

DOCKET NOS. 50-3 and 50-247

Introduction

NRC Generic Letter 83-43, dated December 19, 1983, discussed revisions to notification and reporting requirements in 10 CFR Part 50.72 and Part 50.73 and requested licensees to revise technical specifications to be consistent with the new requirements. By letter dated June 20, 1984, Consolidated Edison Company submitted a request for a proposed amendment to Appendix A of Operating License Nos. DPR-5 amd DPR-26 for Indian Point Units 1 and 2 to accomplish these revisions.

Evaluation

The proposed revisions include changing the definition of "reportable occurrence" to that of "reportable event," deleting unnecessary and conflicting references to reporting requirements in the limiting conditions for operations and surveillance requirements section, and revising the administrative controls section to reference 10 CFR Parts 50.72 and 50.73 and to delete the previous reporting requirements, now unnecessary or conflicting.

The proposed revisions are administrative in nature since they only revise the reporting requirements for reportable events. The revisions do not involve physical changes in plant safety related systems, components, or structures. The revisions will not increase the likelihood of a malfunction of safety related equipment, will not increase the consequences of an accident previously analyzed, nor create the possibility of a malfunction different from those previously evaluated in the Final Safety Analysis Report.

Based on the above, we find the proposed reporting requirement revisions acceptable.

Environmental Consideration

This amendment involves only changes in administrative procedures and requirements. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

Conclusion

We have 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, and (2) 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 or security or to the health and safety of the public.

Dated: April 22, 1985

Principal Contributor

Bernard M. Hillman