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Docket No. 50-247

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. 74 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated April 27, 1981.

The amendment modifies the Technical Specifications to account for the effects that degraded grid voltage may have on plant operations. The amendment contains modifications to your original application. These changes have been discussed with and agreed to by your staff.

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

Sincerely,

John Hannon, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 74 to DPR-26
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
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 April 27, 1981, 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. 74 , 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: December 10, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 74

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
Table 3-1 (Continued)	Table 3-1 (Continued)
Table 3-3 (Continued)	Table 3-3 (Continued)
3.7-2	3.7-2
3.7-6	3.7-6
Table 4.1-1 (Continued)	Table 4.1-1 (Continued)
Table 4.1-1 (Continued)	Table 4.1-1 (Continued)

TABLE 3-1 (Continued)

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

No.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMITS
6.	Steam Generator Water Level (low-low)	Auxiliary Feedwater	$\geq$5% of narrow range instrument span each steam generator
7.	Station Blackout (Undervoltage)	Auxiliary Feedwater	$\geq$40% nominal voltage
8a.	480v Emergency Bus Undervoltage (Loss of Voltage)	--	220V + 100V, -20V 3 sec $\pm$ 1 sec
8b.	480v Emergency Bus Undervoltage (Degraded Voltage)	--	403V $\pm$ 5V 180 sec $\pm$ 30 sec

TABLE 3-3 (Continued)
INSTRUMENTATION OPERATING CONDITION FOR ENGINEERED SAFETY FEATURES

No.	FUNCTIONAL UNIT	1	2	3	4	5
3.	LOSS OF POWER					
	a. 480v Emergency Bus Undervoltage (Loss of Voltage)	2/bus	1/bus	1/bus	0	Cold Shutdown
	b. 480v Emergency Bus Undervoltage (degraded Voltage)	2/bus	2/bus	1/bus	0	Cold Shutdown
4.	AUXILIARY FEEDWATER					
	a. Stm Gen. Water Level-Low-Low					
	i. Start Motor Driven Pumps	3/stm gen	2 in any stm gen.	2 chan. in each stm gen	1	Reduce RCS temperature such that $T < 350^{o}F$
	ii. Start Turbine-Driven Pump	3/stm. gen	2/3 in each of two stm. gen.	2 chan. in each stm. gen.	1	$T < 350^{o}F$
	b. S.I. Start Motor-Driven Pumps	(All safety injection initiating functions and requirements)				
	c. Station Blackout Start Motor-Driven and Turbine-Driven Pumps	2	1	1	0	$T < 350^{o}F$
	d. Trip of Main Feed-water Pumps start Motor-Driven Pumps	2	1	1	0	Hot Shutdown

B. During power operation, [illegible] following components may be i[illegible]erable:

1. Power operation may continue for seven days if one diesel is inoperable provided the 138 kv and the 13.8 kv sources of off-site power are available and the remaining diesel generators are tested daily to ensure operability and the engineered safety features associated with these diesel generator buses are operable.

2. Power operation may continue for 24 hours, if the 138 kv or the 13.8 kv source of power is lost, provided the three diesel generators are operable. This operation may be extended beyond 24 hours provided the failure is reported to the NRC within the subsequent 24-hour period with an outline of the plans for restoration of off-site power.

3. If the 138 KV power source is lost, in addition to satisfying the requirements of specification 3.7.B.2 above, the 6.9 KV bus tie breaker control switches 1-5, 2-5, 3-6, and 4-6 in the CCR shall be placed in the "pull-out" position and tagged to prevent an automatic transfer of the 6.9 KV buses 1,2,3 and 4.

4. One battery may be inoperable for 24 hours provided the other battery and two battery chargers remain operable with one battery charger carrying the dc load of the failed battery's supply system.

C. Gas Turbine Generators:

1. At least one gas turbine generator (GT-1, GT-2 or GT-3) and associated switchgear and breakers shall be operable at all times.

2. A minimum of 54,200 gallons of fuel for the operable gas turbine generator shall be available at all times.

3. If the requirements of 3.7.C.1 or 3.7.C.2 cannot be met, then, within the next seven (7) days, either the inoperable condition shall be corrected or an alternate independent power system shall be established.

4. If the requirements of 3.7.C.3 cannot be satisfied, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.7.C.3 cannot be met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

Conditions of a system-wide blackout could result in a unit trip. Since normal off-site power supplies as required in Specification 3.7.A are not available for startup, it is desirable to be able to blackstart this unit within on-site power supplies as a first step in restoring the system to an operable status and restoring power to customers for essential service. Specification 3.7.D.1 provides for startup using the on-site gas turbine to supply the 6.9 KV loads and the diesels to supply the 480-volt loads. Tie breakers between the 6.9 KV and 480-volt systems are open so that the diesels would not be jeopardized in the event of any incident and would be able to continue to supply 480-volt safeguards power. The scheme consists of starting two reactor coolant pumps, one condensate pump, 2 circulating water pumps and necessary auxiliaries to bring the unit up to approximately 10% power. At this point, loads can be assumed by the main generator and power supplied to the system in an orderly and routine manner.

This Specification (3.7.D.2) is identical with normal start-up requirements as specified in 3.7.A except that off-site power is supplied exclusively from gas turbines with a minimum total power of 37 MW (nameplate rating) which is sufficient to carry out normal plant startup.

As a result of an investigation of the effect components that might become submerged following a LOCA may have on ECCS, containment isolation and other safety-related functions, a fuse and a locked open circuit breaker were provided on the electrical feeder to emergency lighting panel 218 inside containment. With the circuit breaker in the open position, containment electrical penetration H-70 is de-energized during the accident condition. Personnel access to containment may be required during power operation. Since it is highly improbable that a LOCA would occur during this short period of time, the circuit breaker may be closed during that time to provide emergency lighting inside containment for personnel safety.

When the 138 KV source of offsite power is out of service, the automatic transfer of 6.9 KV Buses 1,2,3 and 4 to offsite power after a unit trip could result in overloading of the 20 MVA 13.8 KV/6.9 KV auto-transformer. Accordingly, the intent of specification 3.7.B.3 is to prevent the automatic transfer when only the 13.8 KV source of offsite power is available. However, this specification is not intended to preclude subsequent manual operations or bus transfers once sufficient loads have been stripped to assure that the 20 MVA auto-transformer will not be overloaded by these manual actions.

References

1) FSAR-Section 8.2.1

2) FSAR-Section 8.2.3

TABLE 4.1-1 (CONTINUED)

Channel Description	Check	Calibrate	Test	Remarks
22. Accumulator Level and Pressure	S	R	N.A.	
23. Steam Line Pressure	S	R	M	
24. Turbine First Stage Pressure	S	R	M	
25. Logic Channel Testing	N.A.	N.A.	M	
26. Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	M	
27. Control Room Ventilation	N.A.	N.A.	R	Check damper operation for accident mode with isolation signal
28. Control Rod Protection (for use with LOPAR fuel)	N.A.	R	*	
29. Loss of Power				
a. 480v Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	R	
b. 480v Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	R	
c. 480v Emergency Bus Undervoltage (Alarm)	N.A.	R	M	
30. Auxiliary Feedwater:				
a. Steam Generator Water Level (Low-Low)	N.A.	R	R	

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

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST
b. Station Blackout (Undervoltage)	N.A.	R	R
c. Trip of Main Feed-water Pumps	N.A.	N.A.	R
31. Reactor Coolant System Subcooling Margin Monitor	M	R	N.A.
32. PORV Position Indicator (Limit Switch)	N.A.	R	R
33. PORV Block Valve Position Indicator (Limit Switch)	N.A.	R	R
34. Safety Valve Position Indicator (Acoustic Monitor)	N.A.	R	R
35. Auxiliary Feedwater Flow Rate	N.A.	R	R
36. PORV Actuation/Reclosure Setpoints	N.A.	R	N.A.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 74 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

Introduction and Summary

A request for certain information on the effects that degraded grid voltage may have on plant operations was sent to Consolidated Edison Company (Con-Ed) by the NRC on August 12, 1976. General Design Criterion 17 (GDC 17), "Electric Power Systems," of Appendix A, "General Design Criteria for Nuclear Power Plants," of 10 CFR Part 50 requires that the safety function of each a.c. system shall be to provide sufficient capacity and capability to assure that: (a) specified acceptable fuel design limits and the design conditions for the reactor coolant pressure boundry are not exceeded as a result of anticipated operational occurrences; and (b) the core is cooled and containment integrity and other vital functions are maintained during any of the postulated accidents. GDC 17 further requires that provisions be included to minimize the probability of losing electric power from any one of the remaining supplies as a result of or coincident with the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. A sustained degradation of the offsite power system's voltage could result in the loss of capability of the redundant safety loads, their control circuitry, and the associated electrical components required for performing safety functions. Criteria, staff positions, and proposed technical specifications on degraded grid protection were sent to the licensee by a generic letter dated June 2, 1977. Conformance to the standard technical specifications should provide adequate protection for the degraded grid voltage condition.

Con-Ed's responses were dated Spetember 24, 1976; March 31, 1977; August 29, 1977 (two letters); October 16, 1979; April 28, 1980; August 1, 1980; and April 27, 1981. During our review of their proposed technical specifictions, we found that certain changes were necessary. The licensee agreed to these changes and they have been incorporated. The detailed reviews and technical evaluations of Con-Ed's proposed plant modifications and technical specifications changes were performed by EG&G under contract to the NRC with general supervision provided by the staff. EG&G's Technical Evaluation Report is provided as Attachment 1 to this Safety Evaluation Report. We have reviewed EG&G's Technical Evaluation

Report. Agreement has been reached with Con-Ed on all Technical Specification changes; and all issued are now resolved. We conclude that the proposed electrical design modifications are acceptable.

Evaluation Criteria

The criteria used by EG&G in its technical evaluation of the proposed changes include GDC 17 ("Electrical Power Systems") of Appendix A to 10 CFR 50, IEEE Standard 279-1971 ("Criteria for Protection Systems for Nuclear Power Generating Stations"), ANSI Standard C84.1-1977, (Voltage Ratings for Electrical Power Systems and Equipment -60hz), and staff positions defined in NRC generic letter to Con-Ed dated June 2, 1977.

Proposed Changes, Modifications, and Discussion

The following electrical system design modifications and Technical Specification changes were proposed by Con-Ed.

1. Installation of second level undervoltage relays on each of the four 480 volt class 1E buses. Each bus will have two relays connected in two-out-of two logic; and each relay will be set to operate at 403 volts following a three minute time delay.

2. Change in the time delay settings of the existing first level loss-of-voltage relays from two seconds to three seconds.

3. Addition of undervoltage relays on each 1E bus which will annunciate and alarm in the control room when the bus voltage drops to 93.3% of normal.

4. Additions and changes to the plant Technical Specifications including relay setpoints, time delays, tolerances, testing intervals, and calibration intervals. Con-Ed agreed to monthly testing of the undervoltage alarm relays in lieu of monthly testing of the degraded-grid second level under-voltage relays as written in the model specification. We find this acceptable. By design the alarm relays will always activate prior to operation of the second level degraded-voltage relays which will be calibrated during each refueling interval. Other relatively minor changes to the standard Technical Specifications are acceptable.

Summary

We have reviewed the EG&G Technical Evaluation Report and concur with their findings that:

1. The voltage, time-delay settings, and tolerances of the degraded voltage relays will provide adequate protection for the safety-related loads at all onsite system distribution levels within the expected off-site grid voltage limits.

2. The voltage protection scheme includes coincident logic to preclude spurious trips of the off-site power sources.

3. The time delays selected for relay actuation are adequate.

4. The undervoltage sensing function automatically disconnects the off-site power sources whenever the voltage and relay time delay settings have been exceeded.

5. The voltage monitors and the modifications meet IEEE Standard 279.

   EG&G's Technical Evaluation Report concludes that the proposed 18 month channel test interval does not comply with the NRC staff position as described in our letter of June 2, 1977. This led to further negotiations with Con-Ed which culminated in an agreement to test the alarm relays, rather than the second-level degraded grid relays, each month. We find this to be an acceptable alternative to the method written in the model technical specifications. The model technical specifications were not intended to be absolute requirements. The proposed technical specifications including the testing requirements as now revised are adequate.

Environmental Consideration

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

Conclusion

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

Date: December 10, 1981

EGG-EA-5477

SEPTEMBER 1981

DEGRADED GRID PROTECTION FOR CLASS 1E POWER SYSTEMS, INDIAN POINT NUCLEAR STATION UNIT 2, DOCKET NO. 50-247

D. A. Weber

U.S. Department of Energy

Idaho Operations Office • Idaho National Engineering Laboratory

This is an informal report intended for use as a preliminary or working document

dupe of 8111020372

Prepared for the
U.S. Nuclear Regulatory Commission
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6429

EG&G Idaho

Attachment 1

EG&G Idaho Inc

FORM EG&G-398
(Rev 11-79)

INTERIM REPORT

Accession No. ____________

Report No. EGG-EA-5477

Contract Program or Project Title:

Selected Operating Reactor Issues Program (III)

Subject of this Document:

Degraded Grid Protection for Class 1E Power Systems, Indian Point Nuclear Station Unit 2, Docket No. 50-247

Type of Document:

Technical Evaluation Report

Author(s):

D. A. Weber

Date of Document:

September 1981

Responsible NRC Individual and NRC Office or Division:

P. C. Shemanski, Division of Licensing

This document was prepared primarily for preliminary or internal use. It has not received full review and approval. Since there may be substantive changes, this document should not be considered final

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C.
Under DOE Contract No. DE-AC07-76ID01570
NRC FIN No. A6429

INTERIM REPORT

0425J

DEGRADED GRID PROTECTION FOR CLASS 1E POWER SYSTEMS

INDIAN POINT NUCLEAR STATION UNIT 2

Docket No. 50-247

D. A. Weber
Reliability and Statistics Branch
Engineering Analysis Division
EG&G Idaho, Inc.

September 1981

TAC No. 10028

ABSTRACT

In June 1977, the NRC sent all operating reactors a letter outlining three positions the staff had taken in regard to the onsite emergency power systems. Consolidated Edison Company (Con-Ed) was to assess the susceptibility of the safety-related electrical equipment at the Indian Point Nuclear Station Unit 2, to a sustained voltage degradation of the offsite source and interaction of the offsite and onsite emergency power systems. This report contains an evaluation of Con-Ed's analyses, modifications, and technical specification changes to comply with these NRC positions. The evaluation has determined that Con-Ed does not comply with one of the NRC positions.

FOREWORD

This report is supplied as part of the "Selected Operating Reactor Issues Program (III)" being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Licensing, by EG&G Idaho, Inc., Reliability and Statistics Branch.

The U.S. Nuclear Regulatory Commission funded the work under the authorization, B&R 20 19 01 06, FIN No. A6429.

CONTENTS

TECHNICAL EVALUATION REPORT
DEGRADED GRID PROTECTION FOR CLASS 1E POWER SYSTEMS

INDIAN POINT NUCLEAR STATION UNIT 2

1.0 INTRODUCTION

On June 2, 1977, the NRC requested the Consolidated Edison Company (Con-Ed) to assess the susceptibility of the safety-related electrical equipment at the Indian Point Nuclear Station Unit No. 2 (IP-2) to a sustained voltage degradation of the offsite source and interaction of the offsite and onsite emergency power systems.[1] The letter contained three positions with which the current design of the plant was to be compared. After comparing the current design to the staff positions, Con-Ed was required to either propose modifications to satisfy the positions and criteria or furnish an analysis to substantiate that the existing facility design has equivalent capabilities.

Con-Ed responded to the NRC letter with two submittals dated August 29, 1977.[2,3] These submittals and the submittals of September 20, 1976,[4] September 24, 1976,[5] December 17, 1976,[6] March 31, 1977,[7] June 17, 1977,[8] September 15, 1977,[9] October 16, 1979,[10] April 28, 1980,[11] August 1, 1980,[12] December 31, 1980,[13] April 14, 1981,[14] April 27, 1981,[15] and the Indian Point Unit No. 2 Final Safety Analysis Report (FSAR)[16] complete the information reviewed for this report.

2.0 DESIGN BASE CRITERIA

The design base criteria that were applied in determining the acceptability of the system modifications to protect the safety-related equipment from a sustained degradation of the offsite grid are:

1. General Design Criterion 17 (GDC 17), "Electrical Power Systems," of Appendix A, "General Design Criteria for Nuclear Power Plants," of 10 CFR 50[16]

2. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations"[17]

3. IEEE Standard 308-1974, "Class 1E Power Systems for Nuclear Power Generating Stations"[18]

4. Staff positions as detailed in a letter sent to the licensee, dated June 2, 1977[1]

5. ANSI Standard C84.1-1977, "Voltage Ratings for Electrical Power Systems and Equipment (60 Hz)."[19]

3.0 EVALUATION

This section provides, in Subsection 3.1, a brief description of the existing undervoltage protection at IP-2; in Subsection 3.2, a description of the licensee's proposed modifications for the second-level undervoltage protection; and in Subsection 3.3, a discussion of how the proposed modifications meet the design base criteria.

3.1 Existing Undervoltage Protection. There are four 480V class 1E buses (2A, 3A, 5A, and 6A) for Indian Point 2. Each of the buses is equipped with CV-7 inverse-time relays set at 46% (220V) which automatically strip their associated loads (except safeguard MCC26A and 26B) after 2 seconds. These buses are also equipped with additional CV-7 relays which will initiate load shedding, start the emergency diesel generators, and energize the emergency buses through load sequencing operation.

3.2 Modifications. The licensee has proposed to install two second-level undervoltage relays on each 480 volt safety-related bus in a two-out-of-two logic. The set point for each relay is 403 volts (84%) with a time delay of 180 seconds. The existing time delay on the loss-of-voltage relays has been extended from 120 cycles (2 seconds) to 3 seconds.[16] In addition the licensee has added undervoltage relays on each of the safety-related buses which will provide annunciation to the operator when the bus voltage drops to 93.3%.[10] Proposed changes to the plant's technical specifications were also furnished by the licensee.

3.3 Discussion. The first position of the NRC staff letter[1] required that a second level of undervoltage protection for the onsite power system be provided. The letter stipulates other criteria that the undervoltage protection must meet. Each criterion is restated below, followed by a discussion regarding the licensee's compliance with that criterion.

1. "The selection of voltage and time setpoints shall be determined from an analysis of the voltage requirements of the safety-related loads at all onsite system distribution levels."

   The licensee has provided an analysis of the voltage requirements of the safety-related loads at all onsite system distribution levels and have concluded that the 460V motors are the most limiting safety-related equipment. The analysis was performed for the continously running safety-related motors, all of which have service factors of 1.15 and running load less than the nameplate rating of the motor.

   Con-Ed's proposed Technical Specifications require that the 480V Emergency Bus Undervoltage (Degraded Voltage) relays have a setpoint of 403V $\pm$ 5V. This setpoint and tolerance will provide adequate protection for the safety-related loads at all onsite system distribution levels.

2. "The voltage protection shall include coincident logic to preclude spurious trips of the offsite power sources."

The proposed modification incorporates a two-out-of-two logic scheme, thereby satisfying this criterion.[15]

3. "The time delay selected shall be based on the following conditions:

   a. "The allowable time delay, including margin, shall not exceed the maximum time delay that is assumed in the FSAR accident analysis."

      The proposed maximum time delay of 3 seconds ± 1 second for the loss-of-voltage relays does not exceed this maximum time delay.

   b. "The time delay shall minimize the effect of short-duration disturbances from reducing the unavailability of the offsite power source(s)."

      The licensee's proposed minimum time delay of 180 seconds is long enough to override any short, inconsequential grid disturbances and the starting of large motors.

   c. "The allowable time duration of a degraded voltage condition at all distribution system levels shall not result in failure of safety systems or components."

      The proposed time delay of 180 seconds ± 30 seconds will not result in failure of the safety-related equipment.

4. "The voltage monitors shall automatically initiate the disconnection of offsite power sources whenever the voltage setpoint and time-delay limits have been exceeded."

   A review of the licensee's proposal substantiates that this criterion is met.

5. The voltage monitors shall be designed to satisfy the requirements of IEEE Standard 279-1971."

   The licensee has stated in his proposal that the modifications are designed to meet or exceed IEEE Standard 279.[11]

6. "The technical specifications shall include limiting conditions for operation, surveillance requirements, trip setpoints with minimum and maximum limits, and allowable values for the second-level voltage protection monitors."

The licensee has provided surveillance requirements but the requirement to "test" every 18 months (noted as "R" for refueling in the proposed Technical Specification) is not acceptable. Testing (Channel Functional Test) frequency should agree with the NRC model Technical Specifications (at least once per 31 days).[15]

The second NRC staff position requires that the system design automatically prevent load-shedding of the emergency buses once the onsite sources are supplying power to all sequenced loads. The load-shedding must also be reinstated if the onsite breakers are tripped.

The existing undervoltage relaying scheme for all safety-related buses already has these features incorporated. Only the time delay will be extended, from 2 seconds to 4 seconds when the system is modified for second-level undervoltage protection.

The third NRC staff position requires that certain test requirements be added to the technical specifications. These tests were to demonstrate the full-functional operability and independence of the onsite power sources, and are to be performed at least once per 18 months during shutdown. The tests are to simulate loss of offsite power in conjunction with a safety-injection actuation signal, and to simulate interruption and subsequent reconnection of onsite power sources. These tests verify the proper operation of the load-shed system, the load-shed bypass when the emergency diesel generators are supplying power to their respective buses, and that there is no adverse interaction between the onsite and offsite power sources.

The position is satisfied as the Indian Point 2 Technical Specifications describe tests to demonstrate the full-functional operatility and independence of the onsite systems.

4.0 CONCLUSIONS

Based on the information provided by Con-Ed, it has been determined that the proposed modifications, generally, do not comply with one of the NRC staff positions as described in the NRC letter of June 2, 1977.[1] To comply with this letter the licensee should:

1. Change the unit technical specification surveillance requirements for second-level and loss-of-voltage Channel Functual Test to agree with the NRC requirements (at least once per 31 days).

5.0 REFERENCES

1. NRC letter (R. W. Reid) to Con-Ed, "Staff Positions Relative to the Emergency Power Systems for Operating Reactors," dated June 2, 1977.

2. Con-Ed letter (W. J. Cahill, Jr.) to NRC (R. W. Reid), dated August 29, 1977. (Responding to the NRC's generic letter of August 12, 1976 (Effects of Degraded Grid Voltage) and updating the Con-Ed letter of September 24, 1976.)

3. Con-Ed letter (W. J. Cahill, Jr.) to NRC (R. W. Reid), dated August 29, 1977. (Responding to the NRC letter of June 2, 1977 regarding emergency power systems.)

4. Con-Ed letter (W. J. Cahill, Jr.) to NRC (R. W. Reid), dated September 20, 1976.

5. Con-Ed letter (W. J. Cahill, Jr.) to NRC (R. W. Reid), dated September 24, 1976.

6. Con-Ed letter (W. J. Cahill, Jr.) to NRC (R. W. Reid), dated December 17, 1976.

7. Con-Ed letter (W. J. Cahill, Jr.) to NRC (R. W. Reid), dated March 31, 1977.

8. Con-Ed letter (W. J. Cahill, Jr.) to NRC (R. W. Reid), dated June 17, 1977.

9. Con-Ed letter (W. J. Cahill, Jr.) to NRC (R. W. Reid), dated September 15, 1977.

10. Con-Ed letter (W. J. Cahill, Jr.) to NRC (W. Gamill), dated October 16, 1979.

11. Con-Ed letter (W. J. Cahill, Jr.) to NRC (A. Schwencer), dated April 28, 1980.

12. Con-Ed letter (P. Zarakas) to NRC (S. A. Varga), dated August 1, 1980.

13. Con-Ed letter (J. D. O'Toole) to NRC (S. A. Varga), dated December 31, 1980.

14. Con-Ed letter (J. D. O'Toole) to NRC (S. A. Varga), dated April 14, 1981.

15. Con-Ed letter (J. D. O'Toole) to NRC (S. A. Varga), dated April 27, 1981.

16. Telecon, J. Toma, S. Maskell, NRC, D. Weber, EG&G Idaho, Inc., M. Scott and P. Szabados, Con Ed, August 27, 1981.

17. Final Safety Analysis Report (FSAR) for the Indian Point Nuclear Station Unit 2.

18. General Design Criterion 17, "Electric Power Systems," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

19. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

20. IEEE Standard 308-1974, "Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

21. ANSI C84.1-1977, "Voltage Ratings for Electric Power Systems and Equipment (60 Hz)."

22. IEEE Standard 141-1976, "IEEE Recommended Practice for Electric Power Distribution for Industrial Plants."

23. NEMA Standard, NEMA MG1-1972, "Motors and Generators."

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-247

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 74 to Facility Operating License No. DPR-26, issued to the Consolidated Edison Company of New York, Inc. (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 2 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of the date of issuance.

The amendment modifies the Technical Specifications to account for the effects that degraded grid voltage may have on plant operations.

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 was not required since the amendment does not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendment dated April 27, 1981, (2) Amendment No. 74 to License No. DPR-26, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the White Plains Public Library, 100 Martine Avenue, White Plains, New York. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 10th day of December, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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