ATTACHMENT A

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Technical Specification
Page Revisions

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
Facility Operating License No. DPR-26
August, 1986

TABLE 3.5-2 (Continued)

		1	2	3	4	5
10.	Low Flow Loop $\geq$ 75% F.P.	3/loop	2/loop(any loop)	2/operable loop	1/operable loop	Maintain hot shutdown
	Low Flow Two Loops 10-75% F.P.	3/loop	2/loop(any two loops)	2/operable loop	1/operable loop	
11.	Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop	Maintain hot shutdown
12.	Undervoltage 6.9 KV Bus	1/bus	2	3	2	Maintain hot shutdown
13.	Low frequency 6.9 KV Bus	1/bus	2	3	2	Maintain hot shutdown***
14.	Quadrant power tilt monitors	2	NA	1	0	Log individual upper and lower ion chamber currents once/shift and after load change >10%
15.	Turbine trip (overspeed protection)*****	3	2	2	1	Maintain hot shutdown
16.	Control Rod Protection****	3	2	2	1	During RCS cooldown, manually open reactor trip breakers prior to T_{cold} decreasing below 350°F. Maintain reactor trip breakers open during RCS cooldown when T_{cold} is less than 350°F.
17.	Turbine Trip $\geq$ 35% F.P. A. Low Auto Stop Oil Pressure	3	2	2	1	Maintain reactor power below 35% F.P.
18.	Reactor Trip Logic	2	1	2#	1#	Be in Hot Shutdown within the next forty eight hours

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TABLE 3.5-2 (Continued)

	1	2	3	4	5
19. Reactor Trip Breakers	2	1	2#	1#	With either diverse trip feature inoperable, or the breaker incapable of tripping for any other reason, be in hot shutdown within the next forty eight hours. With a breaker incapable of tripping it shall be bypassed and removed or opened except for the time required for performing maintenance and/or testing to restore it to operability.

F.P. = Rated Power

* If two of four power channels greater than 10% F.P., channels are not required.

** If one of two intermediate range channels greater than 10^{-10} amps, channels are not required.

*** 2/4 trips all four reactor coolant pumps.

**** Required only when control rods are positioned in core locations containing LOPAR fuel.

***** This will provide a turbine trip at all power levels and a reactor trip when greater than or equal to 35% F.P..

\# A reactor trip breaker and/or associated logic channel may be bypassed for maintenance or surveillance testing provided the redundant reactor trip breaker and/or associated logic channel has not been declared inoperable.

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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.	Reactor Trip Logic Channel Testing	N.A.	N.A.	M#	
26.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	M	
27.	Turbine Trip a. Low Auto Stop Oil Pressure	N.A.	R	N.A.	
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)	S	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 opened during RCS cooldown when T_{cold} is less than 350°F.

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TABLE 4.1-1 (CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
42.	Manual Reactor Trip	N.A.	N.A.	R	Includes: 1) Independent verification of reactor trip and bypass breakers undervoltage trip circuit operability up to and including matrix contacts of RT-11/RT-12 from both manual trip initiating devices, 2) independent verification of reactor trip and bypass breaker shunt trip circuit operability through trip actuating devices from both manual trip initiating devices.
43.	Reactor Trip Breaker	N.A.	N.A.	M#	Includes independent verification of undervoltage and shunt trip attachment operability.
44.	Reactor Trip Bypass Breaker	N.A.	N.A.	M#	Includes: 1) Automatic undervoltage trip, 2) Manual shunt trip from either the logic test panel or locally at the switchgear prior to placing breaker into service.

\# Each train shall be tested at least every 62 days on a staggered test basis (i.e., one train per month).

Amendment No.

ATTACHMENT B

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Safety Assessment

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
Facility Operating License No. DPR-26
August, 1986

SAFETY ASSESSMENT

Discussion:

Generic Letter 83-28 was issued by NRC on July 8, 1983 indicating actions to be taken by licensees based on the generic implication of the Salem ATWS events. Item 4.3 of the generic letter requires that modifications be made to improve the reliability of the reactor trip system by implementation of an automatic actuation of the shunt trip attachment on the reactor trip breakers. By letter dated June 14, 1983, the Westinghouse Owners Group (WOG) proposed a generic design modification to implement the automatic shunt trip. By letters dated April 2, 1984 and June 22, 1984, Consolidated Edison provided additional information addressing the plant specific items identified in NRC's August 10, 1983 SER for the WOG's generic shunt trip design. As a condition of the June 22, 1984 SER for the plant specific design, the staff required the submittal of a technical specification change request to require periodic testing of the undervoltage and shunt trip functions and the manual reactor trip switch contacts and wiring following implementation of the modification. A detailed description of the proposed testing was provided by Consolidated Edison letter dated June 22, 1984 and was found acceptable by the staff in NRC's June 22, 1984 SER. On May 23, 1985, NRC issued Generic Letter 85-09 which provided guidance for the preparation of the requested Technical Specification changes. By letter dated February 14, 1986, Consolidated Edison submitted further responses regarding the Technical Specifications and seismic qualification of the automatic shunt trip. In a Supplemental SER transmitted by letter dated June 16, 1986, NRC found Consolidated Edison's response for the seismic qualification issue acceptable and requested that Technical Specification changes responsive to Generic Letter 85-09 be submitted within 60 days of the supplemental SER transmittal date.

The proposed Technical Specification change provides for testing of the undervoltage and shunt trip functions and the manual reactor trip switch contacts and wiring on a refueling frequency as described in our June 22, 1984 letter, for test procedure PT-R51, Revision 1. The proposed Technical Specification change provides for testing of the undervoltage and shunt trip functions on a monthly frequency as described in our June 22, 1984 letter, for test procedure PT-M14A revised to reflect the installation of the automatic shunt trip modification. The proposed Technical Specification revisions are consistent with the guidance contained Generic Letter 85-09.

In addition to the aforementioned Technical Specification changes, two typographical errors were corrected. In amendment No. 107, the first page of Table 3.5-2 was issued as "Table 3.5-2 (1 of 3)" but the subsequent pages were labeled" 3.2 (continued)". The correct label should be "3.5-2 (continued)". The other typographical error was in Table 4.1-1, Item No. 29.a which was written as "400V Emergency Bus Undervoltage". The correct channel description should be "480V Emergency Bus Undervoltage"; similar to items 29.B and 29.c. We do not have a 400V bus.

In Generic Letter 85-09, NRC adopted a graded approach to the reactor trip breaker LCOs, permitting a 48 hour LCO to subcriticality with one of the two diverse trip features inoperable and a six hour LCO with both diverse trip features inoperable or the breaker incapable of tripping for any other reason.

Based on our experience, six hours is insufficient to permit proper repair of an inoperable breaker. Short LCOs can lead to compensating actions which, while fully acceptable, may be less than optimal. Furthermore, recognizing the benefits to breaker reliability from the shunt trip modification, our policy is to immediately bypass the breaker and commence repair activities if one of the diverse trip features were to become inoperable. Our approach would be identical if both diverse trip features were to become inoperable (e.g., immediately bypass the breaker and commence repair activities). Thus our action would be the same whether one or both diverse trip features became inoperable, and therefore the graded LCO serves no useful purpose.

A single LCO of 48 hours is provided for in the requested Technical Specification regardless of whether one or both diverse trip features is inoperable. Since the action taken is the same in both cases (i.e., the reactor trip breaker is bypassed and made inoperable for corrective maintenance regardless of whether one or both diverse trip features was the cause of taking the breaker out of service), the breaker is in fact unavailable for tripping in both cases. If 48 hours is an acceptable LCO with one diverse trip feature inoperable and the breaker out of service, it is also an acceptable LCO with both diverse trip features inoperable and the breaker out of service, because in both cases the breaker is out of service and unavailable for tripping. It is the out-of-service time upon which an LCO must be based, not the number of inoperable diverse trip features. The 48 hour LCO has been selected as the minimum time necessary to repair any breaker problem consistent with maintaining a high degree of overall system availability. The performance reliability of the IP#2 reactor trip breakers has been excellent with no breaker failures recorded to date in over 1000 demands. Accordingly we believe the single 48 hour LCO to be a reasonable and prudent course of action.

Basis for No Significant Hazards Consideration Determination:

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870). Example (ii) of those involving no significant hazards consideration discusses a change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: for example, a more stringent surveillance requirement. The proposed changes to Tables 3.5-2 and 4.1-1 with respect to the reactor trip breakers provide new explicit LCOs and testing requirements consistent with the modified shunt trip design, not previously included in Technical Specifications.

The proposed change does not involve a significant hazards consideration because operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The technical specification changes submitted reflect plant modifications already implemented and reviewed pursuant to 10 CFR 50.59, and as such are expected to enhance the reliability of the reactor trip breakers to trip on demand. The proposed technical specification changes are consistent with guidance contained in Generic Letter 85-09. In addition, the proposed changes constitute additional controls not presently included in the technical specifications. Therefore, this change will not increase the probability or consequences of an accident.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed technical specification changes resulted from extensive review and analysis of the Salem ATWS event and are a result of modifications made as recommended by those analyses. The proposed change would not alter the configuration of any of the plant's safety equipment. Therefore, it has been determined that this change will not create the possibility of a new or different kind of accident from that previously evaluated.

(3) involve a significant reduction in a margin of safety. The modifications made to the plant increase the margin of safety and the proposed technical specifications changes reflect additional conservative administrative controls based on those modifications. Therefore, it has been determined that this change does not involve a significant reduction in a margin of safety.

Therefore, based on the above considerations, and inasmuch as this proposed change is similar to an example for which the Commission has determined no significant hazards considerations exist (i.e., a new limitation or surveillance requirement), we conclude that this proposed change does not constitute a significant hazards consideration.

The proposed changes have been reviewed by Consolidated Edison's Station Nuclear Safety Committee and Nuclear Facilities Safety Committees. Both committees concur that these changes do not represent a significant hazards consideration and will not cause any change in the types or increase in the amounts of effluents or any change in the authorized power level of the facility.