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AUG 24 1981

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. 72 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. This amendment consists of changes to the Operating License and the Technical Specifications in response to your applications transmitted by letters dated March 11, 1981 and April 27, 1981.

The amendment revises the Operating License and the Technical Specifications to incorporate certain of the TMI-2 Lessons Learned Category "A" requirements.

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

Sincerely,

Original signed by:
S. A. Varga

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

RECEIVED AUG 31 1981

Enclosures:
1. Amendment No. 72 to DPR-26
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures;
See next page

Conform language in amendment to Regulatory language of Att. I. WHO

no legal objection

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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. 72
License No. DPR-26

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated March 11, 1981 and April 27, 1981, comply 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 applications, 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, by renumbering paragraph 2.K of Facility Operating License No. DPR-26 to paragraph 3, and by amending paragraph 2.C(2) and adding paragraphs 2.K (inadvertently omitted by Amendment No. 65), 2.L and 2.M to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 72 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

K. The licensee may proceed with and is required to complete the modifications identified in Paragraphs 3.1.1 through 3.1.25 of the NRC's Fire Protection Safety Evaluation (SE) on the facility dated January 31, 1979. The modifications shall be completed in accordance with the schedule contained in Paragraph 3.1 of the SE and supplements thereto.

The licensee is required to implement and maintain the administrative controls identified in Section 6 of the NRC's Fire Protection Safety Evaluation on the Facility dated January 31, 1979. The administrative controls shall be in effect by June 1, 1979.

L. The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

M. The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

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: August 24, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 72

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
iv	iv
3.1-2	3.1-2
------	3.1-2a
3.1-3	3.1-3
3.5-2	3.5-2
3.5-3	3.5-3
------	Table 3-1 (Continued)
Table 3-3	Table 3-3
------	Table 3-3 (Continued)
Table 3-4	Table 3-4
------	Table 3-5
------	Table 3-5 (Continued)
Table 4.1-1 (third sheet)	Table 4.1-1 (third sheet)
------	Table 4.1-1 (fourth sheet)
4.8-1	4.8-1
4.8-2	4.8-2
6-4	6-4

LIST OF TABLES

Frequency Notation	1-1
IP2 Reactor Vessel Core Region Material	3.1-1
Engineered Safety Features Initiation Instrument Setting Limits	3-1
Reactor Trip Instrumentation Limiting Operating Conditions	3-2
Instrumentation Operating Condition for Engineered Safety Features	3-3
Instrumentation Operating Conditions for Isolation Functions	3-4
Table of Indicators and/or Recorders Available to the Operator	3-5
Non-Automatic Containment Isolation Valves Open Continuously or Intermittently for Plant Operation	3.6-1
Safety Related Shock Suppressors (Snubbers)	3.12-1
Fire Detection Instruments	3.13-1
Fire Hose Stations and Fire Hydrants	3.13-2
Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	4.1-1
Frequencies for Sampling Tests	4.1-2
Frequencies for Equipment Tests	4.1-3
Inservice Inspection Requirements for Indian Point No. 2	4.2-1
Containment Isolation Valves	4.4-1
Steam Generator Tube Inspection	4.13-1
Minimum Shift Crew Composition	6.2-1

ΔT trip setpoint for three loop operation has been set in accordance with specification 2.3.1.B-4.

d. Reactor operation with one of the four loops out of service will be permitted for up to 24 hours. If the fourth loop can not be returned to service within 24 hours, the reactor will be put in a hot shutdown condition using normal procedures.

2. Steam Generator

Two steam generators shall be capable of performing their heat transfer function whenever the reactor is critical and the average coolant temperature is above 350°F.

3. Safety Valves

a. At least one pressurizer code safety valve shall be operable, or an opening greater than or equal to the size of one code safety valve flange shall be provided to allow for pressure relief, whenever the reactor head is on the vessel except for hydrostatically testing the RCS in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is critical.

c. The pressurizer code safety valve lift setting shall be set at 2485 psig with $\pm 1\%$ allowance for error.

4. Power Operated Relief Valves (PORVs)/Block Valves

a. Whenever the reactor coolant system is above 350°F, the PORVs and their associated block valves shall be operable with the block valves either open or closed.

b. If a PORV becomes inoperable when above 350°F, its associated block valve shall be maintained in the closed position.

c. If a PORV block valve becomes inoperable when above 350°F, the block valve shall be closed and deenergized.

d. If the requirements of specifications 3.1.A.4.a, 3.1.A.4.b or 3.1.A.4.c above cannot be satisfied, compliance shall be established within one (1) hour, or the reactor shall be placed in the hot shutdown condition within the next six (6) hours and subsequently cooled below 350°F.

5. Pressurizer Heaters

a. Whenever the reactor coolant system is above 350°F, the pressurizer shall be operable with at least 150kw of pressurizer heaters.

b. If the requirements of specification 3.1.A.5.a cannot be met, restore the required pressurizer heater capacity to operable status within 72 hours or place the reactor in hot shutdown within the next 6 hours and subsequently cool below 350°F.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only (1); hence, the specified upper limit of 2% rated power without operating pumps provides a substantial safety factor.

Three loop operation is allowed over a 24 hour period to permit corrective action to return the fourth loop to service and limit the number of unnecessary shutdown cycles. During these periods of three loop operation, the reactor coolant system parameters will be maintained within the limits described for three loop operation in Section 2.1 and 3.1 of the Technical Specifications.

Each of the pressurizer code safety valves is designed to relieve 408,000 lbs. per hr. of saturated steam at the valve set point. Below approximately 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperatures and pressure.[2]

If no residual heat were removed by the Residual Heat Removal System the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for overpressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load[3] without a direct trip or any other control.

Two steam generators capable of performing their heat transfer function will provide sufficient heat removal capability to remove decay heat after a reactor shutdown.

All pressurizer heaters are supplied electrical power from an emergency bus. The requirement that 150kw of pressurizer heaters and their associated controls be operable when the reactor coolant system is above 350°F provides assurance that these heaters will be available and can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVs) can operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to provide a relief path when desirable and to ensure the ability to seal off possible RCS leakage paths. Both the PORVs and the PORV block valves are subject to periodic valve testing for operability in accordance with the ASME Code Section XI as specified in the Indian Point Unit No. 2 Inservice Inspection and Testing Program.

Reference

1) FSAR Section 14.1.6
2) FSAR Section 9.3.1
3) FSAR Section 14.1.10

3.5.5 The cover plate on the rear of the safeguards panel, in the control room, shall not be removed without authorization from the Watch Supervisor.

3.5.6 When the reactor coolant system is above 350°F, the instrumentation requirements as stated in Table 3-5 shall be met.

Basis

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features.(1)(4)

Safety Injection System Actuation

Protection against a Loss of Coolant or Steam Break accident is brought about by automatic actuation of the Safety Injection System which provides emergency cooling and reduction of reactivity.

The Loss of Coolant Accident is characterized by depressurization of the Reactor Coolant System and rapid loss of reactor coolant to the containment. The Engineered Safety Features have been designed to sense the effects of the Loss of Coolant accident by detecting low pressurizer pressure and generate signals actuating the SIS active phase.

The SIS active phase is also actuated by a high containment pressure signal (Hi-Level) brought about by loss of high enthalpy coolant to the containment. This actuation signal acts as a backup to the low pressurizer pressure signal actuation of the SIS and also adds diversity to protection against loss of coolant.

Signals are also provided to actuate the SIS upon sensing the effects of a steam line break accident. Therefore, SIS actuation following a steam line break is designed to occur upon sensing high differential steam pressure between any two steam generators or upon sensing high steam line flow in coincidence with low reactor coolant average temperature or low steam line pressure.

The increase in the extraction of RCS heat following a steam line break results in reactor coolant temperature and pressure reduction. For this reason protection against a steam line break accident is also provided by low pressurizer pressure signals actuating safety injection.

Protection is also provided for a steam line break in the containment by actuation of SIS upon sensing high containment pressure.

SIS actuation injects highly borated fluid into the Reactor Coolant System in order to counter the reactivity insertion brought about by cooldown of the reactor coolant which occurs during a steam line break accident.

Containment Spray

The Engineered Safety Features actuation system also initiate containment spray upon sensing a high containment pressure signal (Hi-Hi Level). The containment spray acts to reduce containment pressure in the event of a loss of coolant or steam line break accident inside the containment. The spray cools the containment directly and limits the release of fission products by absorbing iodine should it be released to the containment.

Containment spray is designed to be actuated at a higher containment pressure (approximately 50% of design containment pressure) than the SIS (2.0 psig). Since spurious actuation of containment spray is to be avoided, it is automatically initiated only on coincidence of Hi-Hi Level containment pressure sensed by both sets of two-out-of-three containment pressure signals.

Steam Line Isolation

Steam line isolation signals are initiated by the Engineered Safety Features closing all steam line stop valves. In the event of a steam line break, this action prevents continuous, uncontrolled steam release from more than

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

Amendment No.

TABLE 3-3

INSTRUMENTATION OPERATING CONDITION FOR ENGINEERED SAFETY FEATURES

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 or 4 CANNOT BE MET
1	SAFETY INJECTION					
a.	Manual	2	1	1	0	Cold Shutdown
b.	High Containment Pressure (Hi Level)	3	2	2	1	Cold Shutdown
c.	High Differential Pressure Between steam Lines	3/steam line	2/steam line	2/steam line	1/steam line	Cold Shutdown
d.	Pressurizer Low Pressure*	3	2	2	1	Cold Shutdown
e.	High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	2/line	1/2 in any 2 lines	1/line in each of 3 lines	2	Cold Shutdown
		4 T_{avg} Signals	2	3	2	
		4 Pressure Signals	2	3	2	
2	CONTAINMENT SPRAY					
a.	Manual	2	1	1	0	Cold Shutdown
b.	High Containment Pressure (Hi Hi Level)	2 sets of 3	2 of 3 in each set	2 per set	1/set	Cold Shutdown

* Permissible bypass if reactor coolant pressure less than 2000 psig.

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

No.	FUNCTIONAL UNIT	1	2	3	4	5
3.	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°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°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°F
	d. Trip of Main Feed-water Pumps start Motor-Driven Pumps	2	1	1	0	Hot Shutdown

TABLE 3-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 or 4 CANNOT BE MET
1.	CONTAINMENT ISOLATION					
a.	Automatic Safety Injection (Phase A)	See Item No. 1 of Table 3-3				Cold Shutdown
b.	Containment Pressure (Phase B)	See Item No. 2 of Table 3-3				Cold Shutdown
c.	Manual					
	Phase A one out of two	2	1	1	0	Cold Shutdown
	Phase B one out of two	2	1	1	0	Cold Shutdown
2.	STEAM LINE ISOLATION					
a.	High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	See Item No. 1(e) of Table 3-3				Cold Shutdown
b.	High Containment Pressure (Hi Hi Level)	See Item No. 2b of Table 3-3				Cold Shutdown
c.	Manual	1/loop	1/loop	1/loop	0	Cold Shutdown
3.	FEEDWATER LINE ISOLATION					
a.	Safety Injection	See Item No. 1 of Table 3-3				
4.	CONTAINMENT PURGE AND PRESSURE RELIEF ISOLATION					
a.	Containment Radioactivity-High (R-11/R-12)	2	1	*	0	*

---

* See Specification 3.1.F.

TABLE 3-5

TABLE OF INDICATORS AND/OR RECORDERS AVAILABLE TO THE OPERATOR

	PARAMETER	1 NO. OF CHANNELS AVAILABLE	2 MIN. NO. OF CHANNELS REQUIRED [(1)]	3 INDICATOR/ RECORDER [(1)]
1.	Pressurizer Water Level	3	2	Indicator/One Channel is recorded
2.	Reactor Coolant System Subcooling Margin Monitor [(2)]	1	1	Indicator
3.	PORV Position Indicator (Limit Switch)	1/Valve	1/Valve	Indicator and alarm
4.	PORV Block Valve Position Indicator (Limit Switch)	1/Valve [(3)]	1/Valve [(3)]	Indicator [(3)]
5.	Safety Valve Position Indicator (Acoustic Monitor)	1/Valve	1/Valve	Indicator
6.	Auxiliary Feedwater Flow Rate	1/S.G.	1/S.G.	Indicator

Footnotes:

(1) Except as specified in another footnote, columns 2 and 3 may be modified to allow the instrument channels to be inoperable for up to 7 days and/or recorder(s) to be inoperable for up to 14 days. If the minimum number of channels required is not restored to meet the above requirements within the time periods specified, then:

a. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start no later than at the end of the specified time period.

b. If the requirements of Columns 2 and 3 are not satisfied within an additional 48 hours, the reactor shall be cooled to below 350°F utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

TABLE 3-5 (Continued)

Footnotes (Cont'd):

(2) If the subcooling margin monitor is inoperable for more than seven (7) days, plant operation may continue for an additional thirty (30) days provide that steam tables are continuously maintained in the control room and the subcooling margin is determined and recorded once a shift.

(3) Except at times when the valve operator is deenergized in accordance with technical specification 3.1.A.4.c.

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 acci dent mode with isolation signa
28. Control Rod Protection (for use with LOPAR fuel)	N.A.	R	*	
29. 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 open during RCS cooldown when T_{cold} is less than 350°F.

TABLE 4.1-1 (Continued)

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
30. Reactor Coolant System Subcooling Margin Monitor	M	R	N.A.
31. PORV Position Indicator (Limit Switch)	M	R	R
32. PORV Block Valve Position Indicator (Limit Switch)	M*	R	R
33. Safety Valve Position Indicator (Acoustic Monitor)	M	R	R
34. Auxiliary Feedwater Flow Rate	M	R	R
35. PORV Actuation/Reclosure Setpoints	N.A.	R	N.A.

* Except when valve operator is deenergized in accordance with specification 3.1.A.4.c.

4.8 AUXILIARY FEEDWATER SYSTEM

Applicability

Applies to periodic testing requirements of the Auxiliary Feedwater System.

Objective

To verify the operability of the Auxiliary Feedwater System and its ability to respond properly when required.

Specification

A. The following surveillance tests shall be performed at refueling intervals:

(1) Verification of proper operation of auxiliary feedwater system components and initiating logic upon receipt of test signals for each mode of automatic initiation.

(2) Verification of the capability of each auxiliary feedwater pump to deliver full flow to the steam generators.

B. The above tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

Basis

The capacity of any one of the three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements. Testing of the auxiliary feedwater system will verify its operability. These specifications establish those surveillance tests to be performed at refueling intervals to verify operability of both the automatic initiation circuitry and the individual components necessary for proper functioning of the auxiliary feedwater system. This testing will verify proper component actuation upon receipt of all required automatic initiation signals and will verify that adequate system flow rates and pressures are obtained with proper valve positioning and pump full flow operation. Both control room instrumentation and visual observation of the equipment will be used to verify proper component operation.

The periodic "operational readiness" testing required by the ASME Code Section XI for pumps and valves in the auxiliary feedwater system is conducted as specified in the Indian Point Unit No. 2 Inservice Inspection and Testing Program and is therefore not included in these specifications.

References

FSAR - Sections 10.4, 14.1.9 and 14.2.5

Table 6.2-1

Minimum Shift Crew Composition**

License Category	During Operations Involving Core Alterations	During Cold Shutdown or Refueling Periods	At All Other Times
Senior Operator License	2*	1	1
Operator License	1	1	1
Non-Licensed	(As Required)	1	2
Shift Technical Advisor (STA)	1	(None Required)	1

*Includes individual with SRO license supervising fuel movement as per Section 6.2.2(e).

**Shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absense of on duty shift crew members provided immediate action is taken to restore the shift crew compostion to within the minimum requirements of Table 6.2-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. DPR-26

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

CONSOLIDATED EDISON COMPANY OF NEW YORK

DOCKET NO. 50-247

Introduction

By letters dated March 11, 1981 and April 27, 1981 Consolidated Edison Company of New York (the licensee) proposed changes to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-26 for Indian Point 2. The changes involve the incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. The licensee's request is in direct response to the NRC staff's letter dated July 2, 1980.

Background Information

By our letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was attached to our letter dated February 21, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-2 Lessons Learned Category "A" items, we requested that licensees amend their TS to incorporate additional Limiting Conditions of Operation and Surveillance Requirements, as appropriate. This request was transmitted to all licensees on July 2, 1980. Included therein were model specifications that we had determined to be acceptable. The licensee's application is in direct response to our request. Each of the issues identified by the NRC staff and the licensee's response is discussed in the Evaluation below.

Evaluation

2.1.1 Emergency Power Supply Requirements

The pressurizer water level indicators, pressurizer relief and block valves, and pressurizer heaters are important in a post-accident situation. Adequate emergency power supplies add assurance of post-accident functioning of these components. The licensee has the requisite emergency power supplies. The licensee has proposed actions in the event of component inoperability. We have reviewed these proposed TSs and find that the emergency power supplies are reasonably ensured for post-accident functioning of the subject components and are thus acceptable.

2.1.3.a Direct Indication of (of Flow) Valve Position

The licensee has provided a direct indication of power-operated relief valve (PORV) and safety valve position in the control room. These indications are a diagnostic aid for the plant operator and provide no automatic action. The licensee has provided TSs with a 31-day channel check and an 18-month channel calibration requirement; thus, the TSs are acceptable and they meet our July 2, 1980 model TS criteria.

2.1.3.b Instrumentation for Inadequate Core Cooling

The licensee has installed a reactor coolant subcooling margin monitor. We previously reviewed this system in our Safety Evaluation dated February 21, 1980. The licensee submitted TSs with a 31-day channel check and an 18-month channel calibration requirement and actions to be taken in the event of component inoperability. We conclude the TSs are acceptable as they meet our July 2, 1980 model TS criteria.

2.1.7.a Auto Initiation of Auxiliary Feedwater Systems

The plant has provisions for the automatic initiation of auxiliary (emergency) feedwater flow on loss of normal feedwater flow. The TSs submitted by the licensee list the appropriate components, describe the tests and provide for proper test frequency. The TSs contain appropriate actions in the event of component inoperability; therefore, we conclude that the TSs are acceptable.

2.1.7.b Auxiliary (Emergency) Feedwater Flow Indication

The licensee has auxiliary (emergency) feedwater flow indication that meets our testability and vital power requirements. The licensee has proposed a TS with 31-day channel check and 18-month channel calibration requirements. We find this TS acceptable as it meets the criteria of our July 2, 1980 model TS criteria.

2.2.1.b Shift Technical Advisor (STA)

Our request indicated that the TSs related to minimum shift manning should be revised to reflect the augmentation of an STA. The licensee's application would add one STA to each shift, except for cold shutdowns and refueling, to perform the function of accident assessment. The STA qualifications have already been approved in Amendment No. 68, dated March 27, 1981. The licensee's application clarifies the requirement in the February 11, 1980 Confirmatory Order that the STA be "on-shift." We find that the licensee's submittal satisfies our requirements and is acceptable.

EVALUATION TO SUPPORT LICENSE CONDITIONS

2.1.4 Integrity of Systems Outside Containment

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to a Systems Integrity Measurements Program. Such a condition would require the licensee to effect an appropriate program to eliminate or prevent the release of significant amounts of radioactivity to the environment via leakage from engineered safety systems and auxiliary systems, which are located outside reactor containment. By letter dated April 27, 1981, the licensee agreed to adopt such a license condition; accordingly we have included this condition in the license.

2.1.8.c Iodine Monitoring

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to iodine monitoring. Such a condition would require the licensee to effect a program which would ensure the capability to determine the airborne iodine concentration in areas requiring personnel access under accident conditions. By letter dated April 27, 1981, the licensee agreed to adopt such a license condition; accordingly, we have included this condition in the license.

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 the amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because that 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: August 24, 1981

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. 72 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 revises the Operating License and the Technical Specifications to incorporate certain of the TMI-2 Lessons Learned Category "A" requirements.

The applications for the amendment comply 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 applications for amendment dated March 11, 1981 and April 27, 1981, (2) Amendment No. 72 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 24th day of August, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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