MARCH 19 1979

Docket No. 50-247

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

Dear Mr. Cahill:

The Commission has issued the enclosed Amendment No. 51 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 letters dated November 2, 1977 and January 6, 1978.

The amendment revises the Technical Specifications concerning an error in the allowable pressurizer heatup rate, definitions of hot shutdown, quadrant power tilt ratio and surveillance intervals, steam generator tube inservice inspection reports, outage times for the boric acid transfer and storage system, inservice inspections and testing (ISI/IST) requirements separately covered by the Indian Point 2 Inservice Inspection and Testing Program, station battery load test intervals, and a number of editorial matters.

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

Sincerely,

Cnst ccp

Original Signed By

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

Happy Valentine's Day & note the 2 changes on p. 2 of SE.

Enclosures:
1. Amendment No. 51 to DPR-26
2. Safety Evaluation
3. Notice of Issuance

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cc: w/enclosures
See next page

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

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A. Schwencer

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Division of Operating Reactors

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2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
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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. 51
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 November 2, 1977, and January 6, 1978, 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, 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. 51, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 19, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 51

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
ii	ii
iv	iv
1-1	101
1-4	104
Table 1	Table 1
3.1-4	3.1-4
3.2-1 thru 3.2-3	3.2-1 thru 3.2-3
3.5-2	3.5-2
Pg 3 of Table 4.1-1	Pg 3 of Table 4.1-1
Pg 2 of Table 4.1-2	Pg 2 of Table 4.1-2
Table 4.1-3	Table 4.1-3
4.3-1	4.3-1
4.5-3 thru 4.5-6	4.5-3 thru 4.5-6
4.6-2	4.6-2
4.6-3	4.6-3
4.13-4	4.13-4
6-15	6-15
6-19	6-19

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LIST OF TABLES

TECHNICAL SPECIFICATIONS

1 DEFINITIONS

The following used terms are defined for uniform interpretation of the specifications.

1.1 a. Rated Power

A steady state reactor thermal power of 2758 MWT.

b. Thermal Power

The total core heat transfer rate from the fuel to the coolant.

1.2 Reactor Operating Conditions

1.2.1 Cold Shutdown Condition

When the reactor is subcritical by at least 1% Δ k/k and T_{avg} is $\leq 200°F$.

1.2.2 Hot Shutdown Condition

When the reactor is subcritical, by an amount greater than or equal to the margin as specified in Technical Specification 3.10 and T_{avg} is > 200°F and $\leq$ 555°F.

1.2.3 Reactor Critical

When the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

1.2.4 Power Operation Condition

When the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

1.8 <u>Quadrant Power Tilt Ratio</u>

The quadrant power tilt ratio shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average. (W-STS)

1.9 <u>Surveillance Intervals</u>

Unless otherwise noted in an individual surveillance requirement, surveillance intervals shall be as specified in Table 1-1 with extensions as provided in 1.10 below. The extensions provided in 1.10 below also apply to surveillance intervals not listed in Table 1-1 unless the extensions are specifically not allowed.

1.10 <u>Surveillance Interval Maximums</u>

Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. A total maximum combined interval time for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval. (W-STS)

Table 1-1

FREQUENCY NOTATION

Notation	Test Frequency/Requirements	Surveillance Interval
Shift S	At least twice per calendar day	N.A.
Daily D	At least once per caldendar day	N.A.
Weekly W	At least once per week	7 days
Monthly M	At least once per month	31 days
Quarterly Q	At least once per three months	92 days
Semi-Annually SA	At least once per six months	6 months
Annually A	At least once per 12 months	12 months
Refueling R	At least once per 18 months	18 months
S/U or P	Prior to each reactor startup	-
N.A.	Not Applicable	-

B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates averaged over one hour (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 for the service period up to 5 effective full-power years. The heatup or cooldown rate shall not exceed 100°F/hr.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b. Figure 3.1-1 and Figure 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The limit lines shown in Figure 3.1-1 and Figure 3.1-2 shall be recalculated periodically using methods discussed in WCAP-7924A and results of surveillance specimen testing as covered in WCAP-7323.[7] The order of specimen removal may be modified based on the results of testing of previously removed specimens. The NRC will be notified in writing as to any deviations from the recommended removal schedule no later than 6 months prior to scheduled specimen removal.

3. The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4. The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3 of the Technical Specifications.

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the Chemical and Volume Control System.

Objective

To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

Specification

A. When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection.

B. The reactor shall not be made critical unless the following Chemical and Volume Control System conditions are met.

1. Two charging pumps shall be operable.

2. Two boric acid transfer pumps shall be operable.

3. The boric acid storage system shall contain a minimum of 4400 gallons of 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) boric acid solution at a temperature of at least 145°F.

4. System piping and valves shall be operable to the extent of establishing one flow path from the boric acid storage system and one flow path from the refueling water storage tank (RWST) to the Reactor Coolant System.

5. Two channels of heat tracing shall be operable for the flow path from the boric acid storage system.

C. During power operation, the requirements of 3.2.B may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.2.B within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.2.B are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operation procedures.

1. One of the two operable charging pumps may be removed from service provided a second charging pump is restored to operable status within 24 hours.

2. One boric acid transfer pump may be out of service provided the pump is restored to operable status within 48 hours.

3. The boric acid storage system may be inoperable provided the RWST is operable and provided that the boric acid storage system is restored to operable status within 48 hours.

4. One channel of heat tracing for the flow path from the boric acid storage system to the Reactor Coolant System may be out of service provided the failed channel is restored to an operable status within 7 days and the redundant channel is demonstrated to be operable daily during that period.

Basis

The Chemical and Volume Control System provides control of the Reactor Coolant System boron inventory. This is normally accomplished by using any one of the three charging pumps in series with either one of the two boric acid transfer pumps. An alternate method of boration will be to use the charging pumps taking suction directly from the refueling water storage tank.

A third method will be to depressurize and use the safety injection pumps. There are three sources of borated water available for injection through 3 different paths.

(1) The boric acid transfer pumps can deliver the contents of the boric acid storage system to the charging pumps.

(2) The charging pumps can take suction from the refueling water storage tank. (2000 ppm boron solution. Reference is made to Technical Specification 3.3A).

(3) The safety injection pumps can take their suctions from either the refueling water storage tank or the boron injection tank.

The quantity of boric acid in storage from either the boric acid storage system or the refueling water storage tank is sufficient to borate the reactor coolant in order to reach cold shutdown at any time during core life.

Approximately 4000 gallons of the 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) of boric acid are required to meet cold shutdown conditions.

Thus, a minimum of 4400 gallons in the boric acid storage system is specified. An upper concentration limit of 13% (22,500 ppm of boron) boric acid in the boric acid storage system is specified to maintain solution solubility at the specified low temperature limit of 145°F. One of two channels of heat tracing is sufficient to maintain the specified low temperature limit.

Reference

FSAR - Section 9.2

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

Basis

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features[1]

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 level and generates signals actuating the SIS active phase based upon the coincidence of these signals. 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 and level 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.

TABLE 4.1-1 (CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
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

TABLE 4.1-2 (Continued)

FREQUENCIES FOR SAMPLING TESTS

FOOTNOTES:

* NA- Not Applicable

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/cc.

(2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 30 minutes making up at least 95% of the total activity of the primary coolant.

(3) $\overline{E}$ determination will be started when the gross analysis indicates $\geq$ 10 μCi/cc and will be redetermined if the primary coolant gross radioactivity changes by more than 10 μCi/cc in accordance with Specification 3.1.D.

(4) When the iodine-131 activity exceeds 10% of the limit in Specification 3.4.A, the sampling frequency shall be increased to a minimum of once each day.

(5) Except as indicated in Specification 3.1.F.7.

TABLE 4.1-3

FREQUENCIES FOR EQUIPMENT TESTS

		Check	Frequency	Maximum Time Between Tests
1.	Control Rods	Rod drop times of all control rods	Each refueling shutdown	**
2.	Control Rods	Partial movement of all control rods	Every 2 weeks during reactor critical operations	20 days
3.	Pressurizer Safety Valves	Set point	Each refueling shutdown	**
4.	Main Steam Safety Valves	Set point	Each refueling shutdown	**
5.	Containment Isolation System	Automatic Actuation	Each refueling shutdown	**
6.	Refueling System Interlocks	Functioning	Each refueling shutdown prior to refueling operation	NA*
7	Primary System Leakage	Evaluate	5 days/week	NA*
8.	Diesel Fuel Supply	Fuel Inventory	Weekly	10 days
9.	Turbine Steam Stop, Control Valves	Closure	Monthly	45 days
10.	Cable Tunnel Ventilation Fans	Functioning	Monthly	45 days
11.	Control Room and Fuel Handling Building Filtration System	Charcoal Filter Pressure Drop Test < 5 inches of water visual inspection Freon - 112 (or equivalent) test $\geq$ 99.5% at ambient conditions	Each refueling shutdown prior to refueling operation***	**

TABLE 4.1-3 (CONTINUED)

FREQUENCIES FOR EQUIPMENT TESTS

	Check	Frequency	Maximum Time Between Tests
12. Containment Air Filtration System	Visual Inspection	Each refueling***	**
	Pressure Drop Test < 5 inches of water	Each refueling***	**
	Charcoal coupons: iodine and ignition temperature 50% removal for methyl iodine, no ignition below 300° C.	Each refueling	**
	HEPA filters DOP ≥ 99% efficiency	Each refueling***	**

* NA - Not Applicable
** See Specification 1.9.
*** Or at any time work on the filters could alter their integrity

4.3 REACTOR COOLANT SYSTEM INTEGRITY TESTING

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To specify tests for Reactor Coolant System integrity after the system is closed following normal opening, modification or repair.

Specification

a) When the Reactor Coolant System is closed after it has been opened, the system will be leak tested at not less than 2335 psig at NDT requirements for temperature.

b) When Reactor Coolant System modifications or repairs have been made which involve new strength welds on components, the new welds shall meet the requirements of the applicable version of ASME Section XI as specified in the Con Edison Inservice Inspection and Testing Program in effect at the time.

c) The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first five (5) effective full-power yrs. of operation. Figure 4.3-1 will be recalculated periodically. Allowable pressures during during cooldown for the leak test temperature shall be in accordance with Figure 3.1-2.

Basis

For normal opening, the integrity of the system, in terms of strength, is unchanged. If the system does not leak at 2335 psig (Operating pressure +100 psi: ±100 psi is normal system pressure fluctuation), it will be leak tight during normal operation.

For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak tightness during normal operation.

4. The above tests will be considered satisfactory if visual observations and control panel indication indicate that all components have operated satisfactorily.

5. Each recombiner air-supply blower shall be started at least at two-month intervals. Acceptable levels of performance shall be that the blowers start, deliver flow, and operate for at least 15 minutes.

D. Containment Air Filtration System

1. Visual inspection of the filter installation shall be performed every 6 months for the first two years and every refueling thereafter, or at any time work on the filters could alter their integrity. In addition, measurement of the pressure drop across the moisture separators and HEPA filters shall be performed at each refueling. Any significant difference in appearance or pressure drop from initial conditions shall be corrected. The acceptance criterion is that the pressure drop across the moisture separator and HEPA filters does not exceed 5.0 inches of water at design flow.

2. The iodine remov.1 efficiency of at least one charcoal filter charcoal coupon from each unit shall be measured every six months for the first two years and every refueling thereafter. The charcoal in the coupons shall be from the same batch as that contained in their respective units. The efficiency shall be measured under the containment conditions representative of the design basis accident (47 psig, 271°F, 100% relative humidity and design basis accident iodine concentration and flow). An ignition temperature test shall be performed at the same time. The acceptance criterion for filter efficiency is 50% for removal of methyl iodine and for ignition test that ignition does not occur at 300°C. If the acceptance criteria are not met, an additional coupon from the unit that failed the tests shall be tested. If the second coupon fails to meet the criteria the charcoal contained in that unit shall be replaced.

3. The charcoal filter isolation valves shall be tested at intervals not greater than once every refueling to verify operability.

4. The HEPA filter banks shall be tested with locally generated DOP* at each refueling shutdown and indications of abnormal leakage corrected. The acceptance criterion is that the value of the efficiencies measured during the test shall be at least 99%.

Basis:

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally inoperative during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes reactor trip, main feedwater isolation and containment isolation, and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is therefore to combine systems tests to be performed during plant refueling shutdowns, with more frequent component tests, which can be performed during reactor operation.

The refueling systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting a test signal is applied to initiate automatic action and verification made that the components receive the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.(1)

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked daily and the initiating circuits are tested monthly (in accordance with Specification 4.1). The testing of the analog channel inputs is accomplished in the same manner as for the reactor protection system. The engineered safety features logic system is tested by means of test switches to simulate inputs from the analog channels. The test switches interrupt the logic matrix output to the master

*Dioctylphthalate particles test

relay to prevent actuation. Verification that the logic is accomplished is indicated by the matrix test light. Upon completion of the logic checks, verification that the circuit from the logic matrices to the master relay is complete is accomplished by use of an ohmmeter to check continuity.

Other systems that are also important to the emergency cooling function are the accumulators, the Component Cooling System, the Service Water System and the containment fan coolers. The accumulators are a passive safeguard. In accordance with Specification 4.1 the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance.

The charcoal portion of the air recirculation system is a passive safeguard which is isolated from the cooling air flow during normal reactor operation. Hence the charcoal should have a long useful lifetime. The filter frames that house the charcoal are stainless steel and should also last indefinitely. However, the visual inspection specified in Section D.1 of this specification will be performed to verify that this is in fact the case. The iodine removal efficiency cannot be measured with the filter cells in place. Therefore at periodic intervals a representative sample of charcoal is to be removed and tested to verify that the efficiencies for removal of methyl iodide, is obtained.[2] The hydrogen recombiner system is an engineered safety feature which will be used only following a loss-of-coolant accident to control the hydrogen evolved in the containment. The system is not expected to be started before about 13 days have elapsed following the accident. At this time the hydrogen concentration in the containment will have reached 2% by volume, which is the design concentration for starting the recombiner system. Actual starting of the system will be based upon containment atmosphere sample analysis. The complete functional tests of each unit at refueling shutdown will demonstrate the proper operation of the recombiner system. More frequent tests of the recombiner control system and air-supply blowers will assure operability of the system. The biannual testing of the containment atmosphere sampling system will demonstrate the availability of this system.

For the four flow distribution valves (856 A, C, D & E), verification of the valve mechanical stop adjustments is performed periodically to provide assurance that the high head safety injection flow distribution is in accordance with flow values assumed in the core cooling analysis.

References

(1) FSAR Section 6.2

(2) FSAR Section 6.4

4. Each diesel generator shall be given a thorough inspection at least annually following the manufacturer's recommendations for this class of stand-by service.

The above tests will be considered satisfactory if the required minimum safeguards equipment operated as designed.

B. Diesel Fuel Tanks

A minimum oil storage of 41,000 gallons will be maintained at the station at all times.

C. Station Batteries (Nos. 21 and 22)

1. Every month the voltage of each cell, the specific gravity and temperature of a pilot cell in each battery and each battery voltage shall be measured and recorded.

2. Every 3 months each battery shall be subjected to a 24 hour equalizing charge, and the specific gravity of each cell, the temperature reading of every fifth cell, the height of electrolyte, and the amount of water added shall be measured and recorded.

3. At each time data is recorded, new data shall be compared with old to detect signs of abuse or deterioration.

4. At each refueling interval, each battery shall be subjected to a load test and a visual inspection of the plates.

Basis

The tests specified are designed to demonstrate that the diesel generators will provide power for operation of equipment. They also assure that the emergency generator system controls and the control systems for the safeguards equipment will function automatically in the event of a loss of all normal 480v AC station service power.

The testing frequency specified will be often enough to identify and correct any mechanical or electrical deficiency before it can result in a system failure. The fuel supply is continuously monitored. An abnormal condition in these systems would be signaled without having to place the diesel generators themselves on test.

Each diesel generator has a continuous rating of 1750 kw with a 2000 hr rating of 2000 kw. Two diesels operating at their continuous rating can power the minimum safeguards loads. A minimum oil storage of 41,000 gallons will provide for operation of the minimum required engineered safeguards on emergency diesel power for a period of 168 hours.

Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails. The periodic equalizing charge will ensure that the ampere-hour capability of the batteries is maintained.

The refueling interval load test for each battery together with the visual inspection of the plates will assure the continued integrity of the batteries. The batteries are of the type that can be visually inspected, and this method of assuring the continued integrity of the battery is proven standard power plant practice.

Reference

FSAR, Section 8.2

4. Interval of Inspection

a. Subsequent to the first inservice inspection of steam generators, completed during the first refueling outage, inservice inspections shall be performed not less than 12 or more than 24 calendar months after the previous inspection.

b. If the results of two consecutive inspections, not including the preservice inspection, all fall in the C-1 category specified in Table 4.13-1, the frequency of inspection may be extended to 40-month intervals. Also, if it can be demonstrated through two consecutive inspections that previously observed degradation has not continued and no additional degradation has occurred, a 40-month inspection interval may be initiated.

B. Corrective Measures

All leaking tubes and defective tubes shall be plugged.

C. Reports

1. Following completion of inservice inspection and plugging of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

2. The complete results of the steam generator tube inservice inspections shall be reported in writing on an annual basis per Specification 6.9.2.f. This report shall include:

   a. Number and extent of tubes inspected.

   b. Location and percent of wall-thickness penetration for each indication of an imperfection.

   c. Identification of tubes plugged.

3. Results of steam generator tube inspections which fall into Category C-3 of Table 4.13-1 require notification of the Commission within 15 days of this determination*. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

Basis

Inservice inspection of steam generators is essential in order to monitor the integrity of the tubing and to maintain surveillance in the event that there is evidence of mechanical damage or progressive

*Note-Table 4.13-1 requires NRC approval prior to startup in one case.

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

Monthly Operating Report

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

6.9.1.6 Each monthly operating report shall include:

a. A tabulation of plant operating data and statistics.

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

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

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

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

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

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Region I Office of Inspection and Enforcement 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. A report covering the X-Y xenon stability tests within three months upon completion of the tests.

c. To provide the Commission with added verifications of the safety and reliability of the pre-pressurized Zircaloy-clad nuclear fuel, a limited program of non-destructive fuel inspections will be conducted. The program shall consist of a visual inspection (e.g., underwater TV, periscope, or other) of the two lead burnup assemblies in each region during the first, second, and third refueling shutdowns. Any condition observed by this inspection which would lead to unacceptable fuel performance may be the object of an expanded surveillance effort. If another domestic plant which contains pre-pressurized fuel of a similar design reaches fuel exposures equal to or greater than at Indian Point Unit No. 2, and if a limited inspection program is or has been performed there, then the program may not have to be performed at Indian Point Unit No. 2. However, such action requires approval of the Nuclear Regulatory Commission. The results of these inspections will be reported to the Nuclear Regulatory Commission.

d. Inoperable fire protection and detection equipment (Specification 3.13).

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 51 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

By letters dated November 2, 1977, and January 6, 1978, Consolidated Edison Company of New York, Inc. (the licensee) requested amendment of the Appendix A Technical Specifications appended to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2 (IP-2). The amendment would make several changes to the Technical Specifications. The proposed changes correct an error in the maximum pressurizer heatup rate, change the definitions of Hot Shutdown Condition and Quadrant Power Tilt Ration, incorporate a new reporting procedure for steam generator tube inspections, change the out of service time for boric acid transfer pumps and storage system, update inspection and test requirements for new welds on components in the reactor coolant system, delete surveillance testing for certain pumps and valves, change the surveillance interval for station batteries, and clarify surveillance intervals. Also proposed are a number of editorial changes. We have made certain revisions to these proposals. These revisions have been discussed with and agreed to by the licensee.

Pressurizer Heatup Rate Error

In August 1977, Mitsubishi Heavy Industries, Ltd. of Japan, noted an inconsistency in the pressurizer heatup rate stated in their Technical Specifications. Specification 3.4.9 required a heatup rate of 200°F/hr; Specification 5.7.1, however, required a heatup rate of 100°F/hr. This discrepancy was reported to the Westinghouse Electric Corporation (Westinghouse), who then reviewed their analysis of the pressurizer heatup rate and determined that the correct heatup rate is 100°F/hr, and that the correct cooldown rate is 200°F/hr; the Technical Specifications for Indian Point 2 allow maximum pressurizer heatup and cooldown rates of 200°F/hr. Westinghouse notified the Nuclear Regulatory Commission (the Commission) and the licensee of this problem. The requested amendment would correct the error in the pressurizer heatup rate limit.

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In designing the pressurizer, Westinghouse performed a thermal stress analysis which analyzed the fatigue resulting from a heatup rate of 100°F/hr and a cooldown rate of 200°F/hr. This analysis meets the standards of the ASME Code, Section III, which requires that the analysis be based on a usage factor. The usage factor represents the fraction of the fatigue life (the total amount of stress that a particular component is designed to handle), with a usage factor of zero implying that no stress has been exerted on the component,and a usage factor of one implying that the stress exerted on the component is equal to the amount of stress that the component is designed to handle. For any piece of equipment, certain components receive more stress than others. For the pressurizer, this component is the surge nozzle, which has a usage factor of 0.9 for the design numbers listed above. This usage factor is such that if the heatup and cooldown rates used in the analysis were exceeded more than a few times, the actual usage factor for the surge nozzle would exceed 1.0, which is not allowable under the ASME Code. Thus, we conclude that reducing the heatup rate limit from 200°F/hr to 100°F/hr is necessary to maintain thermal stresses in the pressurizer to allowable levels. For the same reasons, we further conclude that the cooldown rate limit presently listed in the Technical Specifications is adequate.

Because the current Technical Specifications allow higher rates of pressurizer heatup than the correct limit, it was necessary to determine whether the correct limit had been exceeded in the past. Discussions with Westinghouse indicate that this is unlikely. This is because system capabilities and Technical Specification limits on the rate of reactor coolant system heatup and pressurization effectively preclude pressurizer heatup rates in excess of 50°F to 75°F per hour. Nevertheless, to confirm that this has been the experience at IP-2, we requested that the licensee review all applicable operating records to determine if the limiting heatup rate of 100°F per hour had ever been exceeded. By letters dated August 11, 1977 and August 24, 1977, the licensee reported that there were no instances when the rate of 100°F per hour had been exceeded. Accordingly, we conclude that the only action required for IP-2 is modification of the Technical Specifications to reduce the limiting pressurizer heatup rate from 200°F per hour to 100°F per hour.

On September 25, 1978 Westinghouse was requested to perform an audit review of the stress analyses for components of the reactor coolant pressure boundary to assure that no similar inadvertent error appears in any other portion of the applicable Technical Specifications. By letter dated October 27, 1978 Westinghouse responded by stating that in the past year it had carefully reviewed the stress analysis inputs to the Technical Specifications for five separate plants and, in addition, is completing a very careful, systematic review on the generic June 15, 1978 version of the Westinghouse Standard Technical Specifications and if any further inconsistencies surface during this review process, suitable action would be taken (in the forum of the Westinghouse STS). We find this response acceptable.

Hot Shutdown Definition, Technical Specification 1.2.2

The present definition of hot shutdown uses the wrong mathematical symbol which requires the reactor coolant to be above (not below) a specific T avg value. This definition is also deficient in that it does not cover coolant temperatures above "cold shutdown" as intended. The proposed change to T.S. 1.2.2 acceptably corrects these deficiencies as well as making a minor upward adjustment in the top of the T avg range for this condition.

Elimination of Existing Specification 1.8

The deletion of Specification 1.8 is acceptable since its subject "Reportable Occurrence" is adequately addressed in Specification 6.9.1.7.

Quadrant Power Tilt Ratio

Technical Specification 1.8 implies that the raw detector currents are used for calculating the quadrant power tilt ratio. In actuality, the detector currents are calibrated before the ratio is calculated. Therefore, the proposed change in the definition of quadrant power tilt ratio is acceptable. This proposed definition is the one used in the Standard Technical Specifications for Westinghouse plants (W-STS). This definition should be renumbered Specification 1.8.

Surveillance Intervals

It is recommended that two definitions, 1.9 and 1.10, and a Table, 1.1, be added to Section 1 of the Technical Specifications to clarify surveillance intervals and extension of such intervals. These changes will be consistent with the licensee's proposed change and in addition, have the advantage of being essentially identical to surveillance provisions as found in the Westinghouse Standard Tecnical Specifications (W-STS).

Clarification of Specification 3.2

Technical Specification 3.2 could be interpreted to required that each boric tank shall contain at least 4400.0 gallons of solution. However, to reach cold shutdown conditions, either 4400.0 gallons of boric acid in the boric acid tanks (both tanks added together) or the boric acid in the refueling water storage tank is sufficient. The proposed change to clarify these requirements is acceptable.

Revision of Technical Specification 4.9.6

All tests for the pumps and valves addressed in Technical Specification 4.5 are now included in the "Indian Point Nuclear Generating Unit No. 2 - Inservice Inspection and Testing Program" submitted to the NRC on August 3, 1977, as supplemented on September 22, 1977. The surveillance requirements in the inservice inspection program for these pumps and valves are the same or more restrictive than those presently delineated in the Technical Specifications. Therefore, elimination of these requirements from Technical Specification 4.5 is acceptable since they are appropriately controlled under 10 CFR 50.55(a). Also, the proposed change in Technical Specification 4.3(b) to delete reference to a specific (the 1970) version of ASME Section XI is acceptable on the same basis. The applicable version is specified in the Inservice Inspection and Testing Program.

Miscellaneous Editorial Changes

The changes proposed for Specifications 3.5.5, 6.9.1.6.b, and Table 4.1-2, are basically editorial in nature. The change to Specification 3.5.5 would delete reference to an Annual Operating Report (AOR) (removal of the cover plate on the rear of the safeguard panel). The AOR is no longer used for operations reporting. Also, control of this panel is a matter more appropriate for plant operating procedures. We find this deletion to be acceptable. The change to Specification 6.9.1.6.b would acceptably clarify information required for the Monthly Operating Report and the change to Table 4.1-2 would acceptably correct an editorial oversight that was made in Section 3.1.F when Amendment No. 30 was issued.

Station Battery Load Test Interval

The change to the Technical Specification 4.6.C.4 would require that the station battery load test and visual inspection of the plates be performed at each refueling shutdown. Once a year is presently specified. It is prudent for these tests to be conducted with the reactor shutdown. Since the reactor must be shut down for a refueling operation and, by the proposed definition a refueling interval will be "at least once per 18 months", we find it acceptable to conduct these tests each refueling interval. This 18 month period is also consistent with the Westinghouse Standard Technical Specifications for batteries. In addition, other tests, which have not been changed by this amendment, are already required to be performed on a monthly and quarterly basis. These tests would detect deterioration of the batteries that might occur.

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: Marh 19, 1979

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. 51 to Facility Operating License No. DPR-26, issued to 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 Technical Specifications concerning an error in the allowable pressurizer heatup rate, definitions of hot shutdown, quadrant power tilt ratio, and surveillance intervals, steam generator tube inservice inspection reports, outage times for the boric acid transfer and storage system, inservice inspections and testing (ISI/IST) requirements separately covered by the IP-2 Inservice Inspection and Testing Program, station battery load test intervals, and a number of editorial matters.

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

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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 November 2, 1977 and January 6, 1978; (2) Amendment No. 51 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 Operating Reactors.

Dated at Bethesda, Maryland this 19th day of March, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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