Correction to Amdt. 101 to DPR-26

December 5, 1985

Docket No. 50-247

DISTRIBUTION

Docket File	B. Grimes
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C. Parrish	
OELD	
L. Harmon	
E. Jordan	

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. 104 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated August 2, 1985.

The amendment revises the Technical Specification to delete the Boron Injection Tank (BIT) and its associated limiting conditions for operation and surveillance requirements.

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

Sincerely,

/s/MSlosson

Marylee M. Slosson, Project Manager
PWR Project Directorate #3
Division of PWR Licensing-A

Enclosures:
1. Amendment No.104 to DPR-26
2. Safety Evaluation

cc: w/enclosures
See next page

*SEE PREVIOUS WHITE FOR CONCURRENCES

ORB#1*	PWR#3-A	BC-ORB#1:DL*	OELD*
CParrish	MSlosson:ps	SVarga	MKarman
11/06/85	11/21/85	11/13/85	11/19/85

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Sincerely,

Marylee M. Slosson, Project Manager
Operating Reactors Branch #1
Division of Licensing

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ORB#1:DL*	ORB#1:DL	BC-ORB#1:DL	OELD	AD-OR:DL
CParrish	MSlosson;ps	SVarga	M. Rahman	GLainas
11/06/85	11/8/85	11/ /85	11/19/85	11/ /85

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See next page

ORB#1:DL	ORB#1:DL	BC - ORB#1:DL	OELD	AD:OR:DL
CParrish	MSlosson	SVarga		GLainas
11/6/85	11/ /85	11/ /85	11/ /85	11/ /85

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

CONSOLIDATED EDISION COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.104
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated August 2, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 104 , 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 and is to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Director
PWR Project Directorate #3
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 5, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.2-2	3.2-2
3.3-1	3.3-1
3.3-9	3.3-9
Table 4.1-1 (continued)	Table 4.1-1 (continued)
Table 4.1-2	Table 4.1-2

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.

Bases

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 normally take their suction from the refueling water storage tank.

Amendment No. 104

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the Engineered Safety Features.

Objective

To define those limiting conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, (3) to remove airborne iodine from the containment atmosphere following a Design Basis Accident, (4) to minimize containment leakage to the environment subsequent to a Design Basis Accident.

Specification

The following specifications apply except during low temperature physics tests.

A. Safety Injection and Residual Heat Removal Systems

1. The reactor shall not be made critical, except for low temperature physics tests, unless the following conditions are met:

   a. The refueling water storage tank contains not less than 345,000 gallons of water with a boron concentration of at least 2000 ppm.

   b. Deleted

   c. The four accumulators are pressurized to at least 600 psig and each contains a minimum of 716 ft^3 and a maximum of 731 ft^3 of water with a boron concentration of at least 2000 ppm. None of these four accumulators may be isolated.

   d. Three safety injection pumps together with their associated piping and valves are operable.

   e. Two residual heat removal pumps and heat exchangers together with their associated piping and valves are operable.

   f. Two recirculation pumps together with the associated piping and valves are operable.

1) Assuring with high reliability that the safeguard system will function properly if required to do so.

2) Allowances of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full rated power for at least 100 days, the magnitude of the decay heat decreases after initiating hot shutdown. Thus the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and therefore in such a case the reactor is to be put into the cold shutdown condition.

Valves 1810, 744 and 882 are kept in the open position during plant operation to assure that flow passage from the refueling water storage tank will be available during the injection phase of a loss-of-coolant accident. As an additional assurance of flow passage availability, the valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves to take place. This additional precaution is acceptable since failure to manually re-establish power to close valves 1810 and 882, following the injection phase, is tolerable as a single failure. Valve 744 will not need to be closed following the injection phase. The accumulator isolation valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves from occuring when accumulator core cooling flow is required.

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes.(3) The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of the performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met.(10, 11) The range of core protection as a function of break diameter provided by the various components of the Safety Injection System is presented in Figure 6.2-6 of the FSAR.

TABLE 4.1-1(CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S	N.A.	N.A.	With analog rod position
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A.	
13.	Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14.	Boric Acid Tank Level	W	R	N.A.	Bubbler tube rodded during calibration
15.	Refueling Water Storage Tank Level	W	R	N.A.	
16.	Deleted				
17.	Volume Control Tank Level	N.A.	R	N.A.	
18.	a. Containment Pressure	D	R	M	Wide Range
	b. Containment Pressure	S	R	M	Narrow Range
	c. Containment Pressure(PT-3300,PT-3301)	M	R	N.A.	High Range
19.	Process and Area Radiation Monitoring Systems	D	R	M	
20.	Boric Acid Make-up Flow Channel	N.A.	R	N.A	
21.A	Containment Sump and Recirculation Sump Level (Discrete)	S	M	R	Discrete Level Indication Systems.
21.B	Containment Sump, Recirculation Sump and Reactor Cavity Level (Continuous)	S	R	R	Continuous Level Indication Systems.
21.C	Reactor Cavity Level Alarm	N.A.	R	R	Level Alarm System
21.D	Containment Sump Discharge Flow	S	R	M	Flow Monitor
21.E	Containment Fan Cooler Condensate Flow	S	R	M[6]	[6]Monthly visual inspection of condensate weirs only.

Amendment No. 104

TABLE 4.1-2

FREQUENCIES FOR SAMPLING TESTS

		Check	Frequency	Maximum Time Between Tests
1.	Reactor Coolant Samples	Gross Activity (1) Radiochemical (2) $\bar{E}$ Determination Tritium Activity F, Cl & O_2	5 days/week (1) Monthly Semi-annually (3) Weekly (1) Weekly	3 days 45 days 30 weeks 10 days 10 days
2.	Reactor Coolant Boron	Boron Concentration	Twice/week	5 days
3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Monthly	45 days
4.	Boric Acid Tank	Boron Concentration	Twice/week	5 days
5.	DELETED			
6.	Spray Additive Tank	NaOH Concentration	Monthly	45 days
7.	Accumulator	Boron Concentration	Monthly	45 days
8.	Spent Fuel Pit	Boron Concentration	Prior to Refueling	NA*
9.	Secondary Coolant	Iodine-131	Weekly (4)	10 days
10.	Containment Iodine-Particulate Monitor or Gas Monitor	Iodine-131 and Particulate Activity or Gross Gaseous Activity	Continuous When Above Cold Shutdown(5)	NA

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 104 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

By letters dated August 2, 1985, the Consolidated Edison Company of New York (the licensee) requested an amendment, in the form of changes to the Technical Specifications (TS) to Appendix A of Operating License No. DPR-26 for the Indian Point 2 plant. The proposed changes will eliminate the requirements for a Boron Injection Tank (BIT). Specifically, the licensee intends to delete reference to the BIT, including the limiting condition for operation and the surveillance requirements from Section 3.2 and 3.3, and Tables 4.1-1 and 4.1-2 of the TS. By letter dated October 18, 1985 Consolidated Edison submitted additional supporting documentation.

The BIT provides boric acid solution at 20,000 ppm to the safety injection pump suction header during accident conditions before the Safety Injection System (SIS) takes borated water from the Refueling Water Storage Tank (RWST). Without addition of boric acid from the BIT, the SIS can only inject borated water to the reactor core from the RWST at a reduced boron concentration, which results in a slower cooldown rate. Consequently, elimination of the BIT will affect the containment pressure response for a postulated Main Steam Line Break (MSLB) through changes in the mass and energy release rates.

The licensee has analyzed the consequences of a postulated MSLB. The LOFTRAN computer code was used to calculate the mass and energy releases to the containment for a large double-ended MSLB at full power, with no BIT. The analytical methodology of the LOFTRAN code (Topical Report WCAP-7907) has been previously reviewed and found acceptable by the staff. The licensee calculated the containment pressure response using the methodology described in the FSAR. The total energy release was calculated from the LOFTRAN output. A pressure/energy curve was derived by calculating the total energy for various total pressures, and the steam partial pressure was determined by an iterative calculation. The licensee did not consider containment heat sinks or active heat removal capability in the calculation. The results show a maximum calculated containment pressure of 43 psig, which is below the containment design pressure (47 psig).

The most severe potential steam line break was determined to be inside the containment with the assumption that the reactor coolant pumps did not trip. These conditions maximize the break size by locating it upstream of the steam line flow restrictor and maximize core overcooling. Core overcooling was calculated to add sufficient positive reactivity so that the core returned to a thermal power of 13.6% of rated.

Although the most reactive control rod was assumed to be stuck out, power peaking was limited by voiding in the higher power regions which provided a local negative reactivity feedback. The minimum DNBR remained above the 1.3 limit which the staff utilizes as a threshold for fuel damage.

Although limited clad perforation following a SLB event is permitted by the SRP, the applicant has demonstrated that no clad perforation is predicted to occur.

The staff has reviewed the licensee's scoping analysis and found the assumptions and calculated results to be conservative. Furthermore, since the temperature profile previously approved for equipment qualification, in accordance with staff guidelines, was based on LOCA environmental conditions, the associated temperature response of the containment for the above scoping analysis is not of concern.

Based on a review of the information provided by the licensee and because of the similarity of the licensee's request to other staff actions on boron concentration reduction programs, the staff concludes that the licensee's proposal to eliminate the βIT will not adversely affect the containment functional performance and will present no significant change in the safety margin. Therefore the BIT requirement may be deleted from the Technical Specification.

Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria

for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 5, 1985

Principal Contributors:

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