Docket No. 50-247

September 30, 1985

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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. 98 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 April 10, 1985.

The amendment revises the Technical Specifications to remove the requirement of waiting 400 continuous hours after shutdown before unloading more than one region of fuel assemblies. The amendment permits the discharge of the entire reactor core after a continuous interval of 131 hours following shutdown, the current time constraint for movement of only one region of fuel assemblies.

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/JDNeighbors

Joseph D. Neighbors, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 98 to DPR-26
2. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:DL	ORB#1:DL	ORB#1:DL	BC-ORB#1:DL	OELD	AD/OR:DL
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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. 98
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated April 10, 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. 98 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 30, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.8-2	3.8-2
3.8-3	3.8-3
3.8-5	3.8-5
3.8-6	3.8-6

4. At least one residual heat removal pump and heat exchanger shall be operable.

5. During reactor vessel head removal and while loading and unloading fuel from the reactor, T_{avg} shall be $\leq 140^{o}F$ and the minimum boron concentration sufficient to maintain the reactor subcritical by at least 10% $\Delta k/k$. The required boron concentration shall be verified by chemical analysis daily.

6. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

7. If the spent fuel pit contains spent fuel, the spent fuel cask shall not be moved over <u>any</u> region of the spent fuel pit until the cask handling system has been reviewed by the Nuclear Regulatory Commission and found to be acceptable. Furthermore, any load in excess of the nominal weight of a spent fuel storage rack and associated handling tool shall not be moved on or above El.-95' in the Fuel Storage Building. Additionally, loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool shall not be moved over spent fuel in the spent fuel pit. The weight of installed crane systems shall not be considered part of these loads.

8. The containment vent and purge system, including the radiation monitors which initiate isolation, shall be tested and verified to be operable immediately prior to refueling operations.

9. No movement of fuel in the reactor shall be made until the reactor has been subcritical for at least 131 hours.

10. The minimum water level above the top of reactor pressure vessel flange shall be at least 23 feet (El. 92'0") whenever movement of spent fuel is being made.

11. A dead-load test shall be successfully performed on the fuel storage refueling building crane before fuel movement begins. The load assumed by the refueling crane for this test must be equal to or greater than the maximum load to be assumed by the refueling crane during the refueling operation. A through visual inspection of the refueling crane shall be made after the dead load test and prior to fuel handling.

12. The fuel-handling building charcoal filtration system must be operating whenever spent fuel movement is being made unless the spent fuel has had a continuous 35-day decay period.

13. A licensed senior reactor operator shall be at the site and designated in charge of the operation whenever changes in core geometry are taking place.

the reactor.[2] Periodic checks of refueling water boron concentration ensure the proper shutdown margin. Part 6 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are utilized during refueling to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The 131 hour decay time following plant shutdown and the 23 feet of water above the top of the reactor vessel flanges are consistent with the assumptions used in the dose calculations for fuel-handling accidents both inside and outside of the containment. The analysis of the fuel handling accident inside of the containment is based on an atmospheric dispersion faction ($^{X}/Q$) of 5.1×10^{-4} sec/m^3 and takes no credit for removal of radioactive iodine by charcoal filters. The requirement for the fuel storage building charcoal filtration system to be operating when spent fuel movement is being made provides added assurance that the offsite doses will be within acceptable limits in the event of a fuel-handling accident. The additional month of spent fuel decay time will provide the same assurance that the offsite doses are within acceptable limits and therefore the charcoal filtration system would not be required to be operating.

The presence of a licensed senior reactor operator at the site and designated in charge provides qualified supervision of the refueling operation during changes in core geometry.

References

(1) FSAR - Section 9.5.2

(2) Fuel Densification - Indian Point Nuclear Generating Station Unit No. 2, dated January 1973, Table 3.3.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 98 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 a letter dated April 10, 1985, Consolidated Edison Company of New York (the licensee) proposed the following amendment to the Technical Specifications appended to Facility Operating License No. DPR-26 for Indian Point Nuclear Generating Unit No. 2.

The proposed revision to Technical Specifications deletes the second sentence of the existing Section 3.8 Specification 9, which states "No movement of fuel in the reactor shall be made until the reactor has been subcritical for at least 131 hours. In the event that more than one region of fuel (72 assemblies or less) is to be discharged from the reactor, those assemblies in excess of one region shall not be discharged before a continuous interval of 400 hours has elapsed after shutdown." The revision allows unlimited fuel movement after the reactor has been subcritical for at least 131 hours.

There have been two spent fuel pool expansions (rerackings) from the original capacity of 270 assemblies to 480 assemblies in 1976 and then to 980 assemblies in 1982. This current capacity of 980 assemblies will be sufficient to accommodate the required spent fuel pool storage through August 1993. The licensee proposes no hardware changes or modifications of the spent fuel pool.

The current specification imposed a 400 hour decay waiting time after reactor shutdown prior to making a full core discharge. This revision proposes to reduce the 400 hour time limit to 131 hours for a full core discharge or any unload greater than one core region. The rationale for the proposed change is to provide greater flexibility in the scheduling of refueling activities in future outages and to permit a more expeditious full core discharge should it become necessary due to unforeseen circumstances.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on May 21, 1985 (50 FR 20970). No public comments or request for hearing were received.

Evaluation

The heat load calculations for the discharge of spent fuel to the pool were consistent with the NRC Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling", and included recommended uncertainty factors and actinide contributions. The licensee employed several conservative assumptions for input parameters (Reference 1) as follows:

1. A reactor thermal power of 2813 Mwt and a capacity factor of 100% were used. This is 102% of the rated thermal power of 2758 Mwt.

2. The full core is discharged instantaneously after 131 hours following the shutdown. Nominal discharge time was at least 65 hours.

3. The core is discharged during the hottest season of the year (i.e., river water temperature of 85°F) resulting in the highest initial spent fuel pool temperature prior to the full core discharge.

4. No credit is taken for fuel pool heat loss due to evaporation, convection, conduction or radiation heat transfer mechanisms.

Standard Review Plan, Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup", specifies that for the abnormal maximum heat load (full core unload), the temperature of the pool water must be kept below boiling.

The staff's review determined that the heat load calculations employ the same rationale and methodology as that of the 1976 and 1982 rerackings. The maximum abnormal heat load resulting from a full core discharge after the 131 hour waiting period was calculated to be 3.29x107 Btu/hr (Reference 1), which resulted in a maximum spent fuel pool temperature of 173°F. The actual plant experience during the 1984 refueling outage indicated that the maximum spent fuel pool temperature was approximately 15°F-20°F below the calculated maximum temperature of 173°F.

In the event of a complete failure of the spent fuel pool cooling system, the licensee's calculation shows the maximum normal and abnormal heatup rates as 8.9°F/hr and 16.8°F/hr, respectively. Under these heatup rates, makeup water has to be provided within 8.1 and 2.32 hours respectively to prevent pool boiling (References 1 and 2). The required makeup rates are approximately 39 gpm for the normal heat load and 73 gpm for the abnormal heat load.

The station can provide the makeup water within the above time periods from one of the following: the primary water storage tank, Refueling Water Storage Tank (RWST), and/or fire protection system.

The pool cooling water from the primary water tank can be provided via either one of two 150 gpm capacity makeup pumps. The main fire pump capacity is 1500 gpm and the diesel fire pump capacity is 2500 gpm.

Based on the above, the staff concluded: 1) the heat loads are consistent with the NRC Branch Technical Position ASB 9-2, and 2) the proposed change is more conservative than the Standard Review Plan NUREG-0800, and the Standard Technical Specifications for Westinghouse PWR plants. The staff therefore finds the licensee's proposed change to be acceptable.

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.

References:

1. Licensee (Indian Point Unit 2) Letter dated April 10, 1985 and addressed to Steven A. Varga, Chief, Operating Reactors Branch No. 1, Division of Licensing, NRR

2. Licensee (Indian Point Unit 2) Letter dated July 28, 1981 and addressed to Steven A. Varga, Chief, Operating Reactors Branch No. 1, Division of Licensing, NRR

Dated: September 30, 1985

Principal Contributors:

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