ATTACHMENT A

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Technical Specification
Page Revisions

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

February, 1981

8102090171

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 on 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) and as specified in Specification 4.2.
   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.

follows that the ΔT induced during cooldown results in a calculated higher allowable K_{IR} for finite cooldown rates than for steady state under certain conditions.

Because operation control is on coolant temperature, and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure 3.1-2 represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

Details of these calculations are provided in WCAP-7924A[4].

Pressurizer Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and associated Code Addenda through the Summer 1966 Addendum.

References

(1) Indian Point Unit No. 2 FSAR, Section 4.1.5

(2) ASME Boiler & Pressure Vessel Code, Section III, Summer 1965, N-415.

(3) Indian Point Unit No. 3 FSAR, Section 4.2.5

(4) WCAP-7924A, "Basis for Heatup and Cooldown Limit Curves", W.S. Hazelton, S.L. Anderson, S.E. Yanichko, April 1975.

(5) ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition, Appendix G.

(6) ASTM E185-79, Surveillance Tests on Structural Materials in Nuclear Reactors.

(7) WCAP-7323, "Consolidated Edison Company, Indian Point Unit No. 2 Reactor Vessel Radiation Surveillance Program", S.E. Yanichko, May 1969.

(8) Final Report - SWRI Project 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T", E.B. Norris, June 30, 1977.

ITEM 6.6 (CATEGO[illegible] K-1) - Integrally-Welded Sup[illegible]ts

There are no integrally-welded supports on the valves subject to this examination.

ITEM 6.7 (CATEGORY K-2) - Supports and Hangers

The supports and hangers of the valves subject to this examination shall be visually examined in accordance with Section XI of the code, as shown in Table 4.2-1.

G. Miscellaneous Inspections

ITEM 7.1 - Primary Pump Flywheels

The flywheels shall be visually examined at the first refueling. At each subsequent refueling, one different flywheel shall be examined by ultrasonic methods. The examinations scheduled are shown in Table 4.2-1.

ITEM 7.2 - Materials Irradiation Surveillance Specimens

The reactor vessel surveillance program* includes six specimen capsules to evaluate radiation damage based on pre-irradiation and post-irradiation testing of specimens. The specimens will be removed and examined at the following intervals.

Capsule 1 End of Cycle 1 operation.
Capsule 2 End of Cycle 2 operation.
Capsule 3 End of Cycle 5 operation.
Capsule 4 End of Cycle 8 operation.
Capsule 5 End of Cycle 16 operation.
Capsule 6 Spare

* Refer to FSAR section 4.5, WCAP-7323, and Indian Point Unit No. 2 Application for Amendment to Operating License," sworn to on February 3, 1981.

Amendment No.

TABLE 4.2-1 (sheet 11 of 11)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
6.4	G-1	Pressure-retaining bolting		Not applicable	
6.5	G-2	Pressure-retaining bolting	V	100%	Exception is taken for valves which are not accessible.
6.6	K-1	Integrally-welded supports		Not applicable	
6.7	K-2	Supports and hangers	V	100%	Exception is taken for supports and hangers which are not accessible.
		MISCELLANEOUS INSPECTIONS			
7.1		Primary pump flywheel	V & UT	See Remarks	The flywheels shall be visually examined at the first refueling. At each subsequent refueling, one different flywheel shall be examined by ultrasonic methods.
7.2		Irradiation Specimen Schedule	Tensile and Charpy V Notch (Wedge Open Loading)	See Remarks	Refer to Examination G (Miscellaneous Inspections) Item 7.2 in the bases of this Specification 4.2.

ATTACHMENT B

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Safety Evaluation

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

February, 1981

Safety Evaluation

The proposed changes, contained in Attachment A to this Application, would revise the reactor vessel surveillance specimen removal schedule specified in technical specification sections 3.1.B and 4.2. These changes would update the Indian Point Unit No. 2 surveillance capsule program such that it is consistent with the recent ASTM National Standard E 185-79 recommendations regarding capsule removal schedules and will provide more appropriate data with which to evaluate the effects of radiation on the reactor vessel. In accordance with technical specification 3.1.B.2, the proposed revision has been discussed with and agreed to by the NRC Regulatory Staff and documented by letter dated September 24, 1980 from Mr. Peter Zarakas (Consolidated Edison) to Mr. Steven A. Varga (NRC).

A review of the current version of ASTM E 185-79 indicates that the original surveillance capsule removal schedule for Indian Point Unit No. 2 may result in premature withdrawal of surveillance specimens with respect to obtaining the most apropriate irradiation effects data for the reactor vessel. Based on the present and projected reactor vessel effective full power years and surveillance capsule fluences, the proposed capsule removal schedule will satisfy the criteria of Table 1 of ASTM E 185-79.

The proposed changes do not in any way alter the safety analyses performed for Indian Pint Unit No. 2. The proposed changes have been reviewed by the Station Nuclear Safety Committee and the Consolidated Edison Nuclear Facilities Safety Committee. Both committees concur that these changes do not represent a significant hazards consideration and will not cause any change in the types or an increase in amounts of effluents or any change in the authorized power level of the facility.