ATTACHMENT A

AMENDMENT NO. 1 TO
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

December, 1977

TABLE OF CONTENTS

Amendment No.

TABLE OF CONTENTS (Continued)

LIST OF TABLES

1.6.1 Channel Check

A qualitative determination of acceptable operability by observation of channel behavior during operation. This determination shall include comparison of the channel with other independent channels measuring the same variable.

1.6.2 Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating action.

1.6.3 Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including alarm or trip, and shall be deemed to include the channel functional test.

1.7 Containment Integrity

Containment integrity is defined to exist when:

a. All non-automatic containment isolation valves which are not required to be open during accident conditions, except those required to be open for normal plant operation or testing as identified in Specification 3.6.1, are closed and blind flanges are installed where required.

b. The equipment door is properly closed and sealed by the Weld Channel and Penetration Pressurization System.

c. At least one door in each personnel air lock is properly closed.

d. All automatic containment isolation valves are either operable or in the closed position, or isolated by a closed manual valve or flange that meets the same design criteria as the isolation valve.

e. Containment leakage has been verified in accordance with the surveillance requirements of Specification 4.4, and the requirements of Specification 3.3.D are being satisfied.

requirements of 3.3.B-1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.B-1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

a. Fan cooler unit 23, 24, or 25 may be non-operable during normal reactor operation for a period not to exceed 24 hours, provided both containment spray pumps are demonstrated to be operable.

OR

Fan cooler unit 21 or 22 may be non-operable during normal reactor operation for a period not to exceed 7 days provided both containment spray pumps are demonstrated daily to be operable.

b. One containment spray pump may be out of service during normal reactor operation, for a period not to exceed 24 hours, provided the five fan cooler units are operable and the remaining containment spray pump is demonstrated to be operable.

c. Any valve required for the functioning of the system during and following accident condition may be inoperable provided it is restored to operable status within 24 hours and all valves in the system that provide the duplicate function are demonstrated to be operable.

C. Isolation Valve Seal Water System (IVSWS)

1. The reactor shall not be brought above cold shutdown unless the following requirements are met:

a. The IVSWS shall be operable.

b. The IVSW tank shall be maintained at a minimum pressure of 50 psid and contain a minimum of 144 gallons of water.

2. The requirements of 3.3.C.1 may be modified to allow any one of the following components to be inoperable at any one time:

   a. Any one header of the IVSWS may be inoperable for a period not to exceed seven consecutive days.

   b. Any valve required for the functioning of the system during and following accident conditions provided it is restored to an operable status within seven days and all valves in the system that provide a duplicate function are demonstrated to be operable.

3. If the IVSW System is not restored to an operable status within the time period 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 reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25^{o}F and 100 psi, respectively, over existing values.

   c. In either case, if the IVSW System is not restored to an operable status within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

D. Weld Channel and Penetration Pressurization System (WC & PPS)

1. The reactor shall not be brought above cold shutdown unless:

   (a) All four zones of the WC & PPS are pressurized at or above 47 psig.

(b) The uncorrected air consumption for the WC & PPS is less than or equal to 0.2% of the containment volume per day.

2. The requirements of 3.3.D.1 may be modified as follows:

   a. Any one zone of the WC & PPS may be inoperable for a period not to exceed seven consecutive days.

   b. The uncorrected air consumption for the WC & PPS may be in excess of 0.2% of the containment volume per day for a period not to exceed seven consecutive days.

3. If the WC & PP System is not restored to an operable status within the time period 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 reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values.

   c. In either case, if the WC & PP System is not restored to an operable status within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

The seven day out of service period for the Weld Channel and Penetration Pressurization System and the Isolation Valve Seal Water System is allowed because no credit has been taken for operation of these systems in the calculation of off-site accident doses should an accident occur. No other safeguards systems are dependent on operation of these systems.[13] The minimum pressure settings for the IVSWS and WC & PPS during operation assures effective performance of these systems for the maximum containment calculated peak accident pressure of 47 psig.

References

(1) FSAR Section 9

(2) FSAR Section 6.2

(3) FSAR Section 6.2

(4) FSAR Section 6.3

(5) FSAR Section 14.3.5

(6) FSAR Section 1.2

(7) FSAR Section 8.2

(8) FSAR Section 9.6.1

(9) FSAR Section 14.3

(10) Indian Point Unit No. 2 "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10CFR50.46 and Appendix K of 10CFR50", January 1977.

(11) Letter from William J. Cahill, Jr. of Consolidated Edison Company of New York, to Robert W. Reid of the Nuclear Regulatory Commission, dated July 13, 1976. Indian Point Unit No. 2 Small Break LOCA Analysis.

(12) Indian Point Unit No. 3 FSAR Sections 6.2 and 6.3 and the Safety Evaluation accompanying "Application for Amendment to Operating License" sworn to by Mr. William J. Cahill, Jr. on March 28, 1977.

(13) FSAR Sections 6.5 and 6.6.

3.6 CONTAINMENT SYSTEM

Applicability

Applies to the integrity of reactor containment.

Objective

To define the operating status of the reactor containment for plant operation.

Specification

A. Containment Integrity

1. The containment integrity (as defined in 1.7) shall not be violated unless the reactor is in the cold shutdown condition. However, those non-automatic valves listed in Table 3.6-1 and any test connection valves which are located between containment isolation valves and which are normally closed with threaded caps or blind flanges installed, may be opened if necessary for plant operation or for testing and only as long as necessary to perform the intended function.

2. Non-automatic containment isolation valves may be added to plant systems without prior license amendment to Table 3.6-1 provided that a revision to this Table is included in a subsequent license amendment application.

3. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is sufficient to maintain the shutdown margin $\geq 10\% \frac{\Delta k}{k}$.

4. If containment integrity requirements are not met when the reactor is above cold shutdown, containment integrity shall be restored within four hours or the reactor shall be brought to a cold shutdown condition within the next 36 hours, utilizing normal operating procedures.

B. Internal Pressure

If the internal pressure exceeds 2 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected or the reactor shutdown.

C. Containment Temperature

The reactor shall not be taken above the cold shutdown condition unless the containment ambient temperature is greater than 50°F.

BASIS

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence there would be no pressure buildup in the containment if a Reactor Coolant System rupture were to occur.

The shutdown margins are selected based on the type of activities that are being carried out. The 10% Δk/k shutdown margin when the head is off precludes criticality under any circumstances, even though fuel is being moved. When the reactor head is not to be removed, the specified cold shutdown margin of 1% Δk/k precludes criticality in any occurrence.

Regarding internal pressure limitations, the containment calculated peak accident pressure of 47 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 8 psig.(1) The containment can withstand an internal vacuum of 2.5 psig.(2) The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

The requirement of a 50°F minimum containment ambient temperature is to assure that the minimum service metal temperature of the containment liner is well above the NDT + 30°F criterion for the liner material.(3)

Table 3.6-1 lists non-automatic valves that are designated as part of the containment isolation function. During periods of normal plant operations requiring containment integrity, valves on this Table will be open either continuously or intermittently depending on requirements of the particular protection, safeguards or essential service systems. These valves to be open intermittently are under administrative control and are open only as long as necessary to perform their intended function. In all cases, however, the valves listed in Table 3.6-1 are closed during the post accident period in accordance with plant procedures and consistent with requirements of the related protection, safeguards, or essential service systems.

REFERENCES

(1) FSAR - Section 14.3.5

(2) FSAR - Section 5.5

(3) FSAR - Section 5.1.1.1

TABLE 3.6-1

NON-AUTOMATIC CONTAINMENT ISOLATION VALVES OPEN CONTINUOUSLY

OR INTERMITTENTLY FOR PLANT OPERATION

550	851A	SWN-42	1814A
744	850A	SWN-43	1814B
888A	851B	SWN-44	1814C
888B	850B	SWN-51	1875D (Post Acc.
958	859A	SWN-44	1875E Sample Syst)
959	859C	SWN-51	1875A (")
990C	*891A	SWN-44	1875C (")
1870	*891B	SWN-51	1875F (")
743	*891C	SWN-44	1875B (")
732	*891D	SWN-51	1875G (")
885A	863	SWN-44	1875H (")
885B	1610	SWN-51	1875J (")
205	753H	SWN-71	1882A
226	753G	SWN-71	1882-9
227	SWN-41	SWN-71	1875A (H_2 Supply)
250A	SWN-42	SWN-71	1875-8(")
241A	SWN-43	SWN-71	1876A (")
250B	SWN-41	SA-24	1876-8(")
241B	SWN-42	SA-24	1875B (")
250C	SWN-43	PCV-1111	1875-9(")
241C	SWN-41	PCV-1111	1876B (")
250D	SWN-42	580A	1876-9(")
241D	SWN-43	580B	E-2
869A	SWN-41	UH-43	E-1
878A	SWN-42	UH-44	E-3
869B	SWN-43	990A	E-5
878B	SWN-41	990B	MW-17
			MW-17

* These valves shall be considered deleted from this Table following installation and satisfactory testing of new valve 4312 as part of the Overpressure Protection System modification.

Amendment No.

4.4 CONTAINMENT TESTS

Applicability

Applies to containment leakage.

Objective

To verify that potential leakage from the containment is maintained within acceptable values.

Specification

A. Integrated Leakage Rate

1. Test

a. A full pressure integrated leakage rate test shall be performed at intervals specified in A.3 at the peak accident pressure (P_a) of 47 psig minimum.

b. The test duration shall not be less than 24 hours, and shall be extended a sufficient period of time to verify, by superimposing a known leak rate on the containment, the validity and accuracy of the leakage rate results.

c. A general inspection of the accessible interior and exterior surfaces of the containment structures and components shall be performed prior to performing an integrated leak test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak tightness. If there is evidence of structural deterioration, integrated leakage rate tests shall not be performed until corrective action is taken. Such structural deterioration and corrective actions taken shall be reported as part of the test report.

d. Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves.

2. Acceptance Criteria

The measured leakage rate shall be less than 0.75 L_a where L_a is equal to 0.1 w/o per day of containment steam air atmosphere at 47 psig and 271°F, which are the peak accident pressure and temperature conditions.

3. Frequency

A set of three leakage rate tests shall be performed (during plant shutdown), at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant in service inspection. The start date for the first 10-year service period is September 28, 1973.

B. Sensitive Leakage Rate

1. Test

A sensitive leakage rate test shall be conducted with the containment penetrations, weld channels, and certain double gasketed seals and isolation valve interspaces at a minimum pressure of 47 psig and with the containment building at atmospheric pressure.

2. Acceptance Criteria

The test shall be considered satisfactory if the leak rate for the containment penetrations, weld channel and other pressurized zones is equal to or less than 0.2% of the containment free volume per day.

3. Frequency

A sensitive leakage rate test shall be performed at a frequency of at least every other refueling but in no case at intervals greater than 3 years.

C. Air Lock Tests

1. The containment air locks shall be tested at a minimum pressure of 47 psig and at a frequency of every 6-months. The acceptance criteria is included in D.2.a.

2. Whenever containment integrity is required, verification shall be made of proper repressurization to at least 47 psig of the double-gasket air lock door seal upon closing an air lock door.

D. Containment Isolation Valves

1. Tests and Frequency

a. Isolation valves in Table 4.4-1 shall be tested for operability at intervals no greater than 2 years.

b. Isolation valves in Table 4.4-1 which are pressurized by the Weld Channel and Penetration Pressurization System shall be leakage tested as part of the Weld Channel and Penetration Pressurization System Test at intervals no greater than 2 years.

c. Isolation valves in Table 4.4-1 which are pressurized by the Isolation Valve Seal Water System shall be tested at intervals no greater than 2 years as part of an overall Isolation Valve Seal Water System Test.

d. Isolation valves in Table 4.4-1 which are not pressurized will be tested at intervals no greater than 2 years.

e. Isolation valves in Table 4.4-1 shall be tested with the medium and at the pressure specified therein.

2. Acceptance Criteria

a. The combined leakage rate for the following shall be less than 0.6 L_a: isolation valves listed in Table 4.4-1 subject to gas or nitrogen pressurization testing, air lock testing as specified in C.1, portions of the sensitive leakage rate test described in B.1 which pertain to containment penetrations and double-gasketed seals.

b. The leakage rate into containment for the isolation valves sealed with the service water system shall not exceed 0.36 gpm per fan cooler.

c. The leakage rate for the Isolation Valve Seal Water System shall not exceed 14,700 cc/hr.

3. Containment isolation valves may be added to plant systems without prior license amendment to Table 4.4-1 provided that a revision to this Table is included in a subsequent license amendment application.

E. Containment Modifications

Any major modification or replacement of components of the containment performed after the initial pre-operational leakage rate test shall be followed by either an integrated leakage rate test, or a local leak detection test and shall meet the appropriate acceptance criteria of A.2, B.2, or D.2. Modifications or replacements performed directly prior to the conduct of an integrated leakage rate test shall not require a separate test.

F. Report of Test Results

Each integrated leakage rate test shall be the subject of a summary technical report to be submitted to the Nuclear Regulatory Commission in accordance with the requirements of Appendix J to 10 CFR 50, effective issue date March 16, 1973. Each report shall include leakage test results and a summary analyses of sensitive leak rate, air lock, and containment isolation valve tests performed since the previous integrated leakage rate test.

G. Annual Inspection

A detailed visual examination of the accesible interior and exterior surfaces of the containment structure and its components shall be performed at each refueling shutdown and prior to any integrated leak test, to uncover any evidence of deterioration which may affect either the containment structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, non-destructive tests and inspections, and local testing where practical, prior to the conduct of any integrated leak test. Such repairs shall be reported as part of the test results.

H. Residual Heat Removal System

1. Test

a. (1) The portion of the Residual Heat Removal System that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 350 psig at the interval specified below.

(2) The piping between the residual heat removal pumps suctions and the containment isolation valves in the residual heat removal pump suction line from the containment sump shall be hydrostatically tested at no less than 100 psig at the interval specified below.

b. Visual inspection shall be made for excessive leakage during these tests from components of the system. Any significant leakage shall be measured by collection and weighing or by another equivalent method.

2. Acceptance Criterion

The maximum allowable leakage from the Residual Heat Removal System components located outside of the containment shall not exceed two gallons per hour.

3. Corrective Action

Repairs or isolation shall be made as required to maintain leakage within the acceptance criterion.

4. Test Frequency

Tests of the Residual Heat Removal System shall be conducted at every refueling.

Basis

The containment is designed for a calculated peak accident pressure of 47 psig.(1) While the reactor is operating, the internal environment of the containment will be air at essentially atmospheric pressure and an average maximum temperature of approximately 120°F. With these initial conditions, the temperature of the steam-air mixture at the peak accident pressure of 47 psig is 271°F.

Prior to initial operation, the containment was strength-tested at 54 psig and was leak-tested. The acceptance criterion for this preoperational leakage rate test was established as 0.10 w/o (L_a) per 24 hours at 47 psig and 271°F, which are the peak accident pressure and temperature conditions. This leakage rate is consistent with the construction of the containment,(2) which is equipped with a Weld Channel and Penetration Pressurization System for continuously pressurizing both the penetrations and the channels over all containment liner welds. These channels were independently leak-tested during construction.

The safety analysis has been performed on the basis of a leakage rate of 0.10 w/o per day for 24 hours. With this leakage rate and with minimum containment engineered safeguards operating, the public exposure would be well below 10CFR100 values in the event of the design basis accident.[3]

The performance of a periodic integrated leakage rate test during plant life provides a current assessment of potential leakage from the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, the containment isolation valves are to be closed in the normal manner and without preliminary exercising or adjustments.

The minimum duration of 24 hours for the integrated leakage rate test is established to attain the desired level of accuracy and to allow for daily cyclic variation in temperature and thermal radiation.

The frequency of the periodic integrated leakage rate test is keyed to the schedule for major shutdowns for inservice inspection and refueling. The specified frequency of periodic integrated leakage rate testing is based on the following major considerations.

First is the low probability of leaks in the liner, because of

(a) the tests of the leak-tight integrity of the welds during erection;

(b) conformance of the complete containment to a low leakage rate limit at 47 psig or higher during pre-operational testing; and

(c) absence of any significant stresses in the liner during reactor operation.

Secondly, the Weld Channel and Penetration Pressurization System is in service continuously to monitor leakage from potential leak paths such as the containment personnel lock seals and weld channels, containment penetrations, containment liner weld channels, double-gasketed seals and spaces between certain containment isolation valves and personnel door locks. A leak would be expected to build up slowly and would, therefore, be noted before design limits are exceeded. Remedial action can be taken before the limit is reached.

During normal plant operation, containment personnel lock door seals are continuously pressurized after each closure by the Weld Channel and Penetration Pressurization System. Whenever containment integrity is required, verification is made that seals repressurize properly upon closure of an air lock door.

A full pressure test of the air lock will be periodically performed at 6-month intervals to detect any unanticipated leakage.

The containment isolation valve leakage and sensitive leakage rate measurements obtained periodically, periodic inspection of accessible portions of the containment wall to detect possible damage to the liner plates, combined with the leakage monitoring afforded by the weld Channel and Penetration Pressurization System(4) and IVSWS(5), provide assurance that the containment leakage is within design limits.

The testing of containment isolation valves in Table 4.4-1 either individually or in groups, utilizes the WC & PPS(4) or IVSWS(5) where appropriate and is in accordance with the requirements of Type C tests in Appendix J (issue effective date March 16, 1973) to 10CFR50. The specified test pressures are $\geq$ the peak calculated accident pressure. Sufficient water is available in the Isolation Valve Seal Water System, Primary Water System, Service Water System, Residual Heat Removal System, and the City Water System to assure a sealing function for at least 30 days. The leakage limit for the Isolation Valve Seal Water System is consistent with the design capacity of the Isolation Valve Seal Water supply tank.

The acceptance criterion of 0.6 L_a for the combined leakage of isolation valves subject to gas or nitrogen pressurization, the air lock, containment penetrations and double-gasketed seals is in accordance with Appendix J (issue effective date March 16, 1973) to 10CFR50.

The 350 psig test pressure, achieved either by normal Residual Heat Removal System operation or hydrostatic testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the hydrostatic test pressure for the containment sump return line of 100 psig gives an adequate margin over the highest pressure within the line after a design basis accident. A recirculation system leakage of 2 gal./hr. will limit off-site exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident.

These specifications have been developed using Appendix J (issue effective date March 16, 1973) of 10CFR50 and ANSI N45.4-1972 "Leakage Rate Testing of Containment Structures for Nuclear Reactors" (March 16, 1972) for guidance.

The maximum permissible inleakage rate from the containment isolation valves sealed with service water for the full 12-month period of post accident recirculation without flooding the internal recirculation pumps is 0.36 gpm per fan cooler.

REFERENCES

(1) FSAR - Section 5
(2) FSAR - Section 5.1.7
(3) FSAR - Section 14.3.5
(4) FSAR - Section 6.6
(5) FSAR - Section 6.5

TABLE 4.4-1(Page 1 of 9)

CONTAINMENT ISOLATION VALVES

Valve No.	System [1]	Test Fluid [2]	Minimum Test Pressure (PSIG)
549	PRT to Gas Analyzer	Water [4]	52
548	" " "	Water [4]	52
518	PRT N_2 Supply	Gas	47
550	" " "	Gas	47
552	PRT Makeup Water	Water [4]	52
519	" " "	Water [4]	52
741	RHR return to RCS	Water [5]	52 [3]
744	" " "	Nitrogen [4]	47 [3]
888A	RHR to S.I. Pumps	Nitrogen [4]	47
888B	" " "	Nitrogen [4]	47
958	RHR to Sample System	Nitrogen [4]	47
959	" " "	Nitrogen [4]	47
990C	" " "	Nitrogen [4]	47
1870	RHR from RCS	Nitrogen [4]	47
743	" " "	Nitrogen [4]	47
732	" " "	Nitrogen [4]	47 [3]
885A	Cont. Sump Recirc.Line	Water [5]	52
885B	" " " "	Water [5]	52
201	Letdown Line (CVCS)	Water [4]	52
202	" " " "	Water [4]	52
205	Charging Line (CVCS)	Water [4]	52
226	" " " "	Water [4]	52
227	" " " "	Water [4]	52
250A	RCP Seal Water (CVCS)	Water [4]	52
241A	" " " "	Water [4]	52

Amendment No.

CONTAINMENT ISOLATION VALVES

Valve No.	System (1)	Test Fluid (2)	Minimum Test Pressure (PSIG)
250B	RCP Seal Water (CVCS)	Water (4)	52
241B	" " " "	Water (4)	52
250C	" " " "	Water (4)	52
241C	" " " "	Water (4)	52
250D	" " " "	Water (4)	52
241D	" " " "	Water (4)	52
222	" " " "	Water (4)	52
956E	RCS to Sample System	Water (4)	52
956F	" " " "	Water (4)	52
869A	Cont. Spray System	Water (4)	52
867A	" " " "	Gas	47
878A	" " " "	Gas	47
869B	" " " "	Water (4)	52
867B	" " " "	Gas	47
878B	" " " "	Gas	47
851A	Safety Inj. System	Water (4)	52
850A	" " " "	Water (4)	52
851B	" " " "	Water (4)	52
850B	" " " "	Water (4)	52
859A	S.I. Test Line	Water (4)	52
859C	" " "	Water (4)	52
4312*	Acc. & OPS N_2 Supply	Gas	47
863*	" " " " "	Gas	47

Amendment No.

CONTAINMENT ISOLATION VALVES

Valve No.	System (1)	Test Fluid (2)	Minimum Test Pressure(PSIG)
891A*	Acc. N_2 Supply	Gas	47
891B*	" " "	Gas	47
891C*	" " "	Gas	47
891D*	" " "	Gas	47
956G	Acc. to Sample System	Water (4)	52
956H	" " " "	Water (4)	52
1786	RCDT to Vent Header	Water (4)	52
1787	" " "	Water (4)	52
1610	RCDT N_2 Supply	Gas	47
1616	" " "	Gas	47
1788	RCDT to Gas Analyzer	Water (4)	52
1789	" " "	Water (4)	52
1702	RCDT to WHT (WDS)	Water (4)	52
1705	" " " "	Water (4)	52
797	RCP Comp. Cooling (CCS)	Water (4)	52
784	" " " "	Water (4)	52
FCV-625	" " " "	Water (4)	52
791	Excess Letdown Cool.(CCS)	Water (4)	52
798	" " " "	Water (4)	52
796	" " " "	Water (4)	52
793	" " " "	Water (4)	52
1728	Cont. Sump to WHT (WDS)	Water (4)	52
1723	" " " "	Water (4)	52
1234	Cont. Air Sample	Gas (7)	47
1235	" " "	Gas (7)	47

Amendment No.

CONTAINMENT ISOLATION VALVES

Valve No.	System (1)	Test Fluid (2)	Minimum Test Pressure (PSIG)
1236	Cont. Air Sample	Gas (7)	47
1237	" "	Gas (7)	47
PCV-1229	Air Ejector to Cont.	Gas (7)	47
PCV-1230	" " "	Gas (7)	47
PCV-1214	Steam Gener. Blowdown	Water (4)	52
PCV-1214A	" " "	Water (4)	52
PCV-1215	" " "	Water (4)	52
PCV-1215A	" " "	Water (4)	52
PCV-1216	" " "	Water (4)	52
PCV-1216A	" " "	Water (4)	52
PCV-1217	" " "	Water (4)	52
PCV-1217A	" " "	Water (4)	52
PCV-1223	S.G. to Sample System	Water (4)	52
PCV-1223A	" " "	Water (4)	52
PCV-1224	" " "	Water (4)	52
PCV-1224A	" " "	Water (4)	52
PCV-1225	" " "	Water (4)	52
PCV-1225A	" " "	Water (4)	52
PCV-1226	" " "	Water (4)	52
PCV-1226A	" " "	Water (4)	52
SWN-41	Cont. Fan Cooler-Ser.Wtr.	Water (6)	52
SWN-43	" " " "	Water (6)	52

CONTAINMENT ISOLATION VALVES

Valve No.	System (1)			Test Fluid (2)	Minimum Test Pressure (PSIG)
SWN-42	Cont. Fan	Cooler-Ser.	Wtr.	Water (6)	52
SWN-41	"	"	"	Water (6)	52
SWN-43	"	"	"	Water (6)	52
SWN-42	"	"	"	Water (6)	52
SWN-41	"	"	"	Water (6)	52
SWN-43	"	"	"	Water (6)	52
SWN-42	"	"	"	Water (6)	52
SWN-41	"	"	"	Water (6)	52
SWN-43	"	"	"	Water (6)	52
SWN-42	"	"	"	Water (6)	52
SWN-41	"	"	"	Water (6)	52
SWN-43	"	"	"	Water (6)	52
SWN-42	"	"	"	Water (6)	52
SWN-44	"	"	"	Water (6)	52
SWN-51	"	"	"	Water (6)	52
SWN-44	"	"	"	Water (6)	52
SWN-51	"	"	"	Water (6)	52
SWN-44	"	"	"	Water (6)	52
SWN-51	"	"	"	Water (6)	52
SWN-44	"	"	"	Water (6)	52
SWN-51	"	"	"	Water (6)	52

CONTAINMENT ISOLATION VALVES

Valve No.	System (1)	Test Fluid (2)	Minimum Test Pressure(PSIG)
SWN-44	Cont.Fan Cooler-Ser. Wtr.	Water (6)	52
SWN-51	" " " " "	Water (6)	52
SWN-71	" " " " "	Water (6)	52
SWN-71	" " " " "	Water (6)	52
SWN-71	" " " " "	Water (6)	52
SWN-71	" " " " "	Water (6)	52
SWN-71	" " " " "	Water (6)	52
SA-24	Service Air to Cont.	Water (4)	52
SA-24	" " " "	Water (4)	52
580A	Dead Weight Tester	Gas	47
580B	" " "	Gas	47
UH-43	Auxiliary Steam System	Water (4)	52
UH-44	" " "	Water (4)	52
MW-17	City Wtr. to Cont.	Water (4)	52
MW-17	" " " "	Water (4)	52
1170	Cont. Purge System	Gas (7)	47
1171	" " "	Gas (7)	47
1172	" " "	Gas (7)	47
1173	" " "	Gas (7)	47
1190	Cont. Pressure Relief	Gas (7)	47
1191	" " "	Gas (7)	47
1192	" " "	Gas (7)	47
990A	Recirc. Pump to Samp.Sys.	Nitrogen (4)	47
990B	" " " " "	Nitrogen (4)	47

CONTAINMENT ISOLATION VALVES

Valve No.	System (1)	Test Fluid (2)	Minimum Test Pressure(PSIG)
956A	Pressurizer to Samp. Sys.	Water (4)	52
956B	" " " "	Water (4)	52
956C	" " " "	Water (4)	52
956D	" " " "	Water (4)	52
1814A	Cont. Pressure Instr. Line	Gas	47
1814B	" " " "	Gas	47
1814C	" " " "	Gas	47
1875D	Post Acc. Cont. Sampling	Gas	47
1875E	" " "	Gas	47
1875A	" " "	Gas	47
1875C	" " "	Gas	47
1875F	" " "	Gas	47
1875B	" " "	Gas	47
1875G	" " "	Gas	47
1875H	" " "	Gas	47
1875J	" " "	Gas	47
1882A	O_2 Supply to Cont.	Gas	47
1882-9	" " "	Gas	47
IV-2A	" " "	Gas	47
IV-2B	" " "	Gas	47
1875A	H_2 supp. to H_2 Recomb.	Gas	47
1875-8	" " "	Gas	47
IV-3A	" " "	Gas	47

TABLE 4.4-1 (Page 8 of 9)

CONTAINMENT ISOLATION VALVES

Valve No.	System (1)	Test Fluid (2)	Minimum Test Pressure (PSIG)
1876A	H_2 Supply to H_2 Recomb.	Gas	47
1876-8	" " "	Gas	47
IV-5A	" " "	Gas	47
1875B	" " "	Gas	47
1875-9	" " "	Gas	47
IV-3B	" " "	Gas	47
1876B	" " "	Gas	47
1876-9	" " "	Gas	47
IV-5B	" " "	Gas	47
IA-39	Inst. Air to Cont.	Gas	47
PCV-1228	" " "	Gas	47
E-2	Post Acc.Vent Exhaust Line	Gas (7)	47
E-1	" " " "	Gas (7)	47
E-3	" " " "	Gas (7)	47
E-5	" " " "	Gas (7)	47
85A	Personnel Airlock	Gas	47
85B	" " "	Gas	47
85C	" " "	Gas (7)	47
85D	" " "	Gas (7)	47
95A	Equipment Airlock	Gas	47
95B	" "	Gas	47
95C	" "	Gas (7)	47
95D	" "	Gas (7)	47

Amendment No.

CONTAINMENT ISOLATION VALVES

Notes:

1. System description in which valve is located.
2. Gas Test Fluid indicates either nitrogen or air as test medium.
3. Testable only when at cold shutdown.
4. Isolation Valve Seal Water System.
5. Sealed by Residual Heat Removal System fluid.
6. Sealed by Service Water System.
7. Sealed by Weld Channel and Penetration Pressurization System.

* Following installation and satisfactory testing of new containment isolation valve 4312 as part of the Overpressure Protection System (OPS) modification, valves 891A,B,C & D shall be considered deleted from this Table and valve 4312 shall be considered included in this Table and, therefore, subject to the requirements of specification 4.4.

ATTACHMENT B

AMENDMENT NO. 1 TO 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

December, 1977

Safety Evaluation

On April 16, 1976, an "Application for Amendment to Operating License" was submitted in response to a letter dated August 7, 1975 from Mr. Karl R. Goller to Mr. William J. Cahill, Jr. In that letter, the NRC requested that Consolidated Edison identify planned actions and a schedule to attain conformance with 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors". In subsequent letters from Mr. William J. Cahill, Jr. to Mr. Karl R. Goller dated September 9, 1975 and November 14, 1975 and from Mr. Carl L. Newman to Mr. Karl R. Goller dated April 14, 1976, it was stated that changes to Indian Point Unit No. 2 Technical Specifications would be necessary to meet the requirements of 10 CFR Part 50, Appendix J. Accordingly, the proposed changes to sections 1.7, 3.6 and 4.4 of the Technical Specifications, contained in Attachment A to the April 16, 1976 Application, were submitted to conform the Indian Point Unit No. 2 containment leakage testing program with the requirements of 10 CFR Part 50, Appendix J.

The present "Amendment No. 1 to Application for Amendment to Operating License" is submitted in order to revise and supplement the information provided in the original Application. The proposed changes would revise pages i, ii, iv, 1-4, 3.6-1 through 3.6-3, 4.4-2 through 4.4-10, and Table 3.6-1 as proposed in the April 16, 1976 Application to reflect typographical corrections, editorial changes and revisions effected by recently issued license amendments. In addition, the proposed changes contained on pages i, 3.3-4, 3.3-4(a), 3.3-4(b), and 3.3-15 would revise section 3.3 to incorporate additional limiting conditions for operation (LCOs) regarding the operability of the Isolation Valve Seal Water System (IVSWS) and the Weld Channel and Penetration Pressurization System (WC & PPS).

The revised requirements would allow any one header of the IVSWS and any one zone of the WC & PPS to be inoperable for a maximum of seven (7) consecutive days during power operation. This would permit testing or maintenance to be performed on these systems without removing the unit from service. This maximum 7 day out-of-service period for the IVSWS and the WC & PPS is permissible since no credit has been taken for these systems in the calculation of offsite accident doses and no other safeguards systems are dependent on the operation of these systems. The proposed revisions to section 3.3 of the Indian Point Unit No. 2 Technical Specifications have been modeled after the corresponding sections of the Indian Point Unit No. 3 Technical Specifications and no modifications to the plant are required as a result of these changes.

The new revised section 4.4.D.1, contained in Attachment A to this Amendment, would now render the technical specification requirement for testing of containment isolation valves consistent with the requirements of Appendix J to Part 50 of the Commission's regulations, which allows a two year interval between surveillance tests of containment isolation valves.

The proposed revisions to Tables 3.6-1 and 4.4-1 would reflect the addition of hardware to be installed inside containment for the Overpressure Protection System (OPS) during the upcoming refueling outage. As part of this modification, a new check valve 4312 will be installed in series with existing valve 863 to provide containment isolation for the new Overpressure Protection System. Check valve 4312 will be leak tested and qualified in accordance with Appendix J requirements. This check valve will also satisfy containment isolation requirements presently satisfied by valves 891A, B, C and D for the N_2 supply to the ECCS accumulators. The proposed changes would permit deletion of valves 891A, B, C and D from Tables 3.6-1 and 4.4-1 and addition of check valve 4312 to Table 4.4-1 following installation and satisfactory testing of valve 4312.

In addition, to clarify the use of Tables 3.6-1 and 4.4-1, it has been explicitly stated in the specification for each table that containment isolation valves may be added without prior License Amendment. A revision to the table would be included in a subsequent License Amendment Application. Valves cannot be deleted from these tables without prior License Amendment.

Since many of the proposed technical specification page revisions, contained in Attachment A to the April 16, 1976 Application, have been modified by this present "Amendment No. 1 to Application for Amendment to Operating License", the revised pages provided in Attachment A to the original Application are superseded in total by the page revisions contained in Attachment A to this Amendment. This provides a complete, unified and updated package of proposed Indian Point Unit No. 2 technical specification revisions to effect compliance with the requirements of Appendix J to 10 CFR Part 50.

The proposed changes do not in any way alter the safety analyses performed for Indian Point Unit No. 2. The proposed changes have been reviewed by the Station Nuclear Safety Committee and the 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 of or increase in the amounts of effluents or any change in the authorized power level of the facility.