Docket File
50-247

REGULATORY DOCKET FILE COPY

AUGUST 28 1980

Docket No. 50-247

Mr. William J. Cahill, Jr.
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. 63 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 letter dated April 16, 1976, as revised by letters dated January 6, 1978 and December 31, 1979.

The amendment makes changes to the Technical Specifications related to primary containment leakage testing. In connection with this action, the Commission has granted an exemption which allows the licensee to verify the airlock door seals following each use by pressurizing between the double-gasketed seals to a pressure of Pa. This is an exemption from the portion of Paragraph III.D.2 of Appendix J to 10 CFR 50 which states: "However, airlocks which are opened during such intervals, shall be tested after each opening."

We have reviewed the proposed airlock testing and associated acceptance criteria and find that it adequately demonstrates the leakage integrity of the airlock. We find that granting the proposed exemption from the requirements of Appendix J is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest.

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

Sincerely,

Original signed by
Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

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Enclosures:
1. Amendment No. 63 to DPR-26
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 28, 1980

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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. 63
License No. DPR-26

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

   A. Pursuant to 10 CFR 50.12 of the Commission's Regulations, the Commission has authorized an exemption from the requirements of Appendix J to 10 CFR Part 50;

   B. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated April 16, 1976, as revised January 6, 1978 and December 31, 1979, 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;

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

   D. 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;

   E. 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

   F. 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 63 , 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

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 28, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 63

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
i	i
ii	ii
iv	iv
1-3	1-3
3.3-4	3.3-4
	3.3-4(a)
	3.3-4(b)
3.3-15	3.3-15
3.6-1	3.6-1
3.6-2	3.6-2
	3.6-3
	Table 3.6-1
4.4-1 through 4.4-9	4.4-1 through 4.4-9
	Table 4.4-1 (8 sheets)

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Amendment No. 63, Unit 2

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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 52 psig 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 have been 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°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. <u>Weld Channel and Penetration Pressurization System (WC & PPS)</u>

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

   (a) All required portions of the four WC & PPS zones 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 $25^{o}F$ 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", December 1978.

(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% $\Delta k/k$ 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% $\Delta k/k$ 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.

<u>REFERENCES</u>

(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	878B	SWN-44	1814B
744	851A	SWN-51	1814C
888A	850A	SWN-44-1	1875D
888B	851B	SWN-51-1	1875E
958	850B	SWN-44-2	1875A
959	859A	SWN-51-2	1875C
990C	859C	SWN-44-3	1875F
1870	863	SWN-51-3	1875B
743	1610	SWN-44-4	1875G
732	753H	SWN-51-4	1875H
885A	753G	SWN-71	1875J
885B	SWN-41	SWN-71-1	1882A
205	SWN-42	SWN-71-2	1882-D
226	SWN-43	SWN-71-3	4429
227	SWN-41-1	SWN-71-4	1875-K
250A	SWN-42-1	SA-24	4430
241A	SWN-43-1	SA-24-1	1876-C
250B	SWN-41-2	PCV-1111-1	4431
241B	SWN-42-2	PCV-1111-2	1875-L
250C	SWN-43-2	580A	4432
241C	SWN-41-3	580B	1876-D
250D	SWN-42-3	UH-43	E-2
241D	SWN-43-3	UH-44	E-1
869A	SWN-41-4	990A	E-3
878A	SWN-42-4	990B	E-5
869B	SWN-43-4	1814A	MW-17
			MW-17

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^oF, 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.

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 every refueling but in no case at intervals 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 every refueling but in no case at intervals 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 every refueling but in no case at intervals 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 every refueling but in no case at intervals 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 pursuant to specification 6.9.2.a and 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. Visual Inspection

A detailed visual examination of the accessible 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 8)

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

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
956G	Acc. to Sample System	Water (4)	52
956H	" " " "	Water (4)	52

CONTAINMENT ISOLATION VALVES

Valve No.	System(1)	Test Fluid(2)	Minimum Test Pressure (PSIG)
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
1236	" " "	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

TABLE 4.4-1 (Page 4 of 8)

CONTAINMENT ISOLATION VALVES

Valve No.	System (1)	Test Fluid (2)	Minimum Test Pressure (PSIG)
PCV-1215A	Steam Gener. Blowdown	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
SWN-42	" " " "	Water (6)	52
SWN-41-1	" " " "	Water (6)	52
SWN-43-1	" " " "	Water (6)	52
SWN-42-1	" " " "	Water (6)	52
SWN-41-2	" " " "	Water (6)	52
SWN-43-2	" " " "	Water (6)	52
SWN-42-2	" " " "	Water (6)	52
SWN-41-3	" " " "	Water (6)	52
SWN-43-3	" " " "	Water (6)	52
SWN-42-3	" " " "	Water (6)	52

CONTAINMENT ISOLATION VALVES

Valve No.	System[1]	Test Fluid[2]	Minimum Test Pressure (PSIG)
SWN-41-4	Cont. Fan Cooler-Ser.Wtr.	Water (6)	52
SWN-43-4	" " " "	Water (6)	52
SWN-42-4	" " " "	Water (6)	52
SWN-44	" " " "	Water (6)	52
SWN-51	" " " "	Water (6)	52
SWN-44-1	" " " "	Water (6)	52
SWN-51-1	" " " "	Water (6)	52
SWN-44-2	" " " "	Water (6)	52
SWN-51-2	" " " "	Water (6)	52
SWN-44-3	" " " "	Water (6)	52
SWN-51-3	" " " "	Water (6)	52
SWN-44-4	" " " "	Water (6)	52
SWN-51-4	" " " "	Water (6)	52
SWN-71	" " " "	Water (6)	52
SWN-71-1	" " " "	Water (6)	52
SWN-71-2	" " " "	Water (6)	52
SWN-71-3	" " " "	Water (6)	52
SWN-71-4	" " " "	Water (6)	52
SA-24	Service Air to Cont.	Water (4)	52
SA-24-1	" " "	Water (4)	52
580A	Dead Weight Tester	Gas	47
580A	" " "	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

CONTAINMENT ISOLATION VALVES

Valve No.	System[1]	Test Fluid[2]	Minimum Test Pressure (PSIG)
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
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

CONTAINMENT ISOLATION VALVES

Valve No.	System (1)	Test Fluid (2)	Minimum Test Pressure (PSIG)
1882A	O_2 Supply to Cont.	Gas	47
1882-D	" " "	Gas	47
IV-2A	" " "	Gas	47
IV-2B	" " "	Gas	47
4429	H_2 Supp. to H_2 Recomb.	Gas	47
1875-K	" " " " "	Gas	47
IV-3A	" " " " "	Gas	47
4430	" " " " "	Gas	47
1876-C	" " " " "	Gas	47
IV-5A	" " " " "	Gas	47
4431	" " " " "	Gas	47
1875-L	" " " " "	Gas	47
IV-3B	" " " " "	Gas	47
4432	" " " " "	Gas	47
1876-D	" " " " "	Gas	47
IV-5B	" " " " "	Gas	47
IA-39	Inst. Air to Cont.	Gas	47
PCV-1228	" " "	Gas	47
E-2	Post Acc.Vent Exh. 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

CONTAINMENT ISOLATION VALVES

Valve No.	System (1)		Test Fluid (2)	Minimum Test Pressure (PSIG)
85D	Personnel Airlock		Gas (7)	47
95A	Equipment Airlock		Gas	47
95B	"	"	Gas	47
95C	"	"	Gas (7)	47
95D	"	"	Gas (7)	47

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 63 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

On August 7, 1975 (Reference 1), the NRC requested Consolidated Edison Company to review its containment leakage testing program for Indian Point, Unit 2 and the associated Technical Specifications, for compliance with the requirements of Appendix J to 10 CFR Part 50.

Appendix J to 10 CFR Part 50 was published on February 14, 1972. Since by this date there were already many operating nuclear plants and a number more in advanced stages of design or construction, the NRC decided to have these plants re-evaluated against the requirements of this new regulation. Therefore, beginning in August 1975, requests for review of the extent of compliance with the requirements of Appendix J were made of each licensee. Following the initial responses to these requests, NRC staff positions were developed which would assure that the objectives of the testing requirements of the above cited regulation were satisfied. These staff positions have since been applied in our review of the submittals filed by the Indian Point, Unit 2 licensee. The results of our evaluation are provided below.

Evaluation

Our consultant, the Franklin Research Center, has reviewed the licensee's submittals (References 2, 3, 4, and 5) and prepared the attached evaluation of containment tests for Indian Point, Unit 2. We have reviewed this evaluation and concur in its bases and findings.

In its report, the staff's consultant recommended that the proposed Technical Specifications (T.S.) 4.4.D be modified to require that the Type C test for containment isolation valves and associated systems be performed during each reactor shutdown for refueling but in no case at intervals greater than two years. The proposed specification would simply require Type C tests at intervals no greater than 2 years. We have discussed the above described modification with the licensee and the licensee has agreed to adopt the changes.

On December 31, 1979, the licensee submitted Amendment 2 to Application for Amendment to Operating License (Reference 6), in which additional changes to the Technical Specifications were proposed (as compared to those changes proposed in Amendment 1 to Application for Amendment to Operating License) (Reference 5). The proposed changes in Amendment 2 concerning containment tests that were verbally identified by the licensee include:

1. the deletion of, "The start date for the first 10-year service period in September 28, 1973," from the proposed T.S. 4.4.A.3;

2. the addition of, "pursuant to specification 6.9.2.a," to the proposed T.S. 4.4.F;

3. the correction from "Annual Inspection" to "Visual Inspection" in the proposed T.S. 4.4.G; and

4. the renumbering of certain valves in Table 4.4-1.

In addition, Section 3.3.D.1(a) has been modified to eliminate a potential inconsistency with Section 1.7.c. Section 1.7.c requires that at least one door in each personnel air lock be closed to satisfy containment integrity. We agree that only one door need be closed to satisfy containment integrity and therefore find this change acceptable.

Based on our review of the enclosed evaluation report as prepared by our consultant and on the above discussion, the following conclusions are made regarding the Appendix J review for Indian Point, Unit 2:

1. The licensee's request for excluding valves 753H, 753G, MOV-822A, MOV-822B and PCV111 from Type C testing are found to be acceptable. No exemption from Appendix J requirement is necessary.

2. The proposed T.S. 4.4 (Containment Tests) submitted by the licensee in Amendment 2 to the Application for Amendment to Operating License (Reference 6) is found to be acceptable along with a minor modification to the proposed T.S. 4.4.D.

3. Proposed T. S. 4.4.B requires an exemption from the current requirements of Appendix J to permit airlock door seals after each use in lieu of a complete airlock test. This exemption is found to be acceptable.

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: August 29, 1980

References

[1] NRC Generic Letter from Mr. Karl Goller, Acting Director for Operating Reactors, to Consolidated Edison Company of New York, Inc., (CEC) dated August 7, 1975.

[2] CEC letter from Mr. W. J. Cahill, Jr., Vice President to Mr. Karl Goller, Acting Director for Operating Reactors, dated September 9, 1975.

[3] CEC letter from Mr. C. L. Newman, Vice President to Mr. Karl Goller, Assistant Director, DOR, dated April 14, 1976.

[4] LeBouef, Lamb, Leiby, and MacRay letter to Mr. Ben Rusche, Director, NRR, dated April 16, 1976, which forwarded an Application for Amendment to Facility Operating License DPR-26, dated April 14, 1976.

[5] LeBouef, Lamb, Lieby, and MacRay letter of January 6, 1978, which forwarded Amendment 1 to the Application for Amendment to Facility Operating License DPR-26, dated December 1977.

[6] CEC letter dated December 31, 1979, which forwarded Amendment 2 to the Application for Amendment to Facility Operating License DPR-26, dated December 1979.

Franklin Research Center
A Division of The Franklin Institute

April 30, 1980

United States Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. E. J. Butcher, Jr.
Project Officer

Reference: FRC Project C5257
NRC Contract NRC-03-79-118
NRC TAC No. 10912
FRC Task No. 24
Title: Technical Evaluation Report, Containment Leakage Testing, Indian Point Unit 2

Dear Mr. Butcher:

Attached is the Technical Evaluation Report of the submittal of the Consolidated Edison Company of New York, Inc. relating to Containment Leakage Testing for Indian Point Unit 2.

This completes FRC's efforts on FRC Task No. 24.

Very truly yours,

S P Carfagno

S. P. Carfagno
Project Manager

SPC/TDG/dg

cc: E. G. Adensam
Y. S. Huang

Attach.

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TECHNICAL EVALUATION REPORT

CONTAINMENT LEAKAGE RATE TESTING

INDIAN POINT UNIT 2

NRC DOCKET NO. 50-247

NRC TAC NO. 10912

NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257

FRC TASK 24

Prepared by

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Lead NRC Engineer: Y. S. Huang

APRIL 25, 1980

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TECHNICAL EVALUATION REPORT

CONTAINMENT LEAKAGE TESTING
INDIAN POINT UNIT 2

CONTENTS

TECHNICAL EVALUATION REPORT

CONTAINMENT LEAKAGE TESTING

INDIAN POINT UNIT 2

1.0 BACKGROUND

On August 7, 1975 [1], the NRC requested Consolidated Edison Company of New York, Inc. (CEC) to review the containment leakage testing program for Indian Point Unit 2 (IP-2) and to provide a plan for achieving full compliance where necessary, including appropriate design modifications, changes to technical specifications, and requests for exemption from the requirements pursuant to 10CFR50.12.

On September 9, 1975 [2], CEC replied that modifications to the IP-2 technical specifications would be necessary. On April 14, 1976 [3], CEC submitted a request for exclusion of six valves from the Type C testing requirements of Appendix J. On April 16, 1976 [4], CEC submitted an Application for Amendment to Operating License DPR-26. Subsequently, on December 16, 1976 [5], CEC submitted Amendment 1 to Application for Amendment to Operating License DPR-26 which modified and supplemented the submittal of Reference [4].

This report provides a technical evaluation as to the acceptability of the requests for exemption from the requirements of Appendix J submitted by CEC in Reference [3] and also provides a technical evaluation of the proposed technical specification changes submitted by References [4] and [5] as they relate to the containment leakage testing requirements of Appendix J.

2.0 REVIEW CRITERIA

The criteria by which the technical evaluation was conducted included 10CFR50, Appendix J, Containment Leakage Testing, and ANSI N45.4-1972, Leakage Rate Testing of Containment Structures for Nuclear Reactors. Where applied to the following evaluations, the criteria are either referenced or are briefly stated where necessary to support the result of the evaluations. Furthermore, in recognition of the plant-specific conditions which could lead to requests for exemption not explicitly covered by the regulations, the NRC directed that the technical review constantly emphasize safety aspects and the basic intent of Appendix J that potential containment atmospheric leakage paths be identified, monitored and maintained below established limits.

3.0 TECHNICAL EVALUATION

3.1 REQUESTS FOR EXEMPTION

Reference [3] requested exemption from the requirements of Appendix J in order to exclude four valves in the component cooling water (CCW) system and two valves in the weld channel and penetration pressurization system (WC and PPS) from the containment leakage testing program. The following sections provide an evaluation of these requests.

3.1.1 Component Cooling Water Supply and Return Lines to the Recirculation Pump Motors

CEC requested that Valves 753H and 753G (CCW supply and return line isolation valves to the recirculation pump motors) be excluded from testing in accordance with Appendix J. CEC's basis for this request is that during both normal plant operation and post-accident conditions, these manually operated valves are in the open position to provide component cooling water to the recirculation pumps located within the containment. CEC also stated that the portion of the CCW system within the containment is a closed system and that the portion of the CCW system outside containment supplying cooling water to the pumps is also a closed system and monitored for radioactivity. In view of the above, CEC concluded that a Type C test of these valves, as required by Appendix J, would serve no particular purpose.

Evaluation. The requirements for Type C testing of containment isolation valves are prescribed by Sections II.H and III.A.1.(d) of Appendix J. Based upon CEC's statement that Valves 753H and 753G are normally open manual valves in a closed system inside containment which is designed for continuous operation during both normal and emergency conditions (and therefore not liable to rupture as result of a LOCA), Type C testing is not required by either Section II.H or III.A.1.(d). Consequently, FRC finds that the proposed exclusion from Type C testing for Valves 753H and 753G is acceptable because the regulation does not require testing. No exemption from the requirements of Appendix J is necessary.

3.1.2 Component Cooling Water Return Lines from the Residual Heat Exchangers

CEC requested that Valves MOV-822A and MOV-822B (CCW return line isolation valves from the residual heat exchangers) be excluded from testing in accordance with Appendix J. CEC's basis for the request is that these valves are closed during plant operation and open during accident conditions (after receiving an opening signal). CEC further stated that the portion of the CCW system piping inside containment is a closed system and that the portion of the CCW system outside containment supplying cooling water to the residual heat exchangers is also a closed system and monitored for radioactivity. CEC concluded that a Type C test of these valves, as required by Appendix J, would serve no particular purpose.

Evaluation. The requirements for Type C testing of containment isolation valves are prescribed by Sections II.H and III.A.1.(d) of Appendix J. Furthermore, Section II.B defines a containment isolation valve as one relied upon to perform a containment isolation function.

These valves are in a closed system inside containment which is designed to remain operational throughout the post-accident period (and therefore not liable to rupture as result of a LOCA). Furthermore, the design of the system is such that even if one of the isolation valves fails to open on signal or were to fail shut after opening, the shut valve is always pressurized with water thorugh the common return header. Consequently, regardless of the position of the valves or other possible single active failures within the CCW system, Valves MOV-882A and MOV-822B are not relied upon to perform a containment isolation function since no path exists for the leakage of containment atmosphere and therefore Appendix J does not required that they be tested.

FRC finds that the proposed exclusion from Type C testing for Valves MOV-882A and MOV-882B is acceptable because the regulation does not require testing. No exemption from the requirements of Appendix J is necessary.

3.1.3 Air Supply Line to the Weld Channel and Penetration Pressurization System

CEC requested that Valves PCV-1111 (two valves, one in each penetration supplying air to the WC and PPS) be excluded from testing in accordance with Appendix J. CEC's basis for this request is that these manual isolation valves are required to be in the open position to assure continuous pressurization of the system. CEC further stated that the system is considered to be a closed system inside containment and that the supply of air to Valves PCV-1111 is always higher than peak accident pressure within containment and therefore, no potential exists for leakage from the contianment through these valves. CEC concluded that Type C testing for these valves would serve no purpose.

Evaluation. The requirements for Type C testing of containment isolation valves are prescribed by Sections II.H and III.A.1.(d) of Appendix J. These sections do not require Type C testing of manual valves in closed systems inside containment which are not liable to rupture as result of a LOCA.

Review of the IP-2 FSAR for the WC and PPS indicates that the system is designed to engineered-safety-feature-system criteria (and therefore not liable to rupture as result of a LOCA), that it is a closed system inside containment, and that it has sufficient redundancy in the supply of air to provide air at pressures in excess of peak calculated accident pressure throughout the accident. This review confirmed CEC's contention that there is no potential for leakage of containment atmosphere through these isolation valves. The review also revealed that Appendix J does not require that these valves be tested.

Therefore, FRC finds that the exclusion of Valves PCV-1111 (two valves, one in each penetration supplying air to the WC and PPS) from Type C testing is acceptable because the regulation does not require that they be tested. No exemption from the requirements of Appendix J is necessary.

3.2 PROPOSED TECHNICAL SPECIFICATION CHANGES

Reference [4] forwarded an Application for Amendment to Facility Operating License DPR-26. Reference [5] forwarded Amendment 1 to that application. Only those portions of Amendment 1 which are applicable to this evaluation have been considered in this report.

3.2.1 Containment Tests (Specification 4.4)

CEC's purpose in providing the proposed revision to this part of the technical specification (TS) was to revise the TS in accordance with the requirements of Appendix J. The FRC's evaluation of each major subsection of TS 4.4 is presented below.

3.2.1.1 Integrated Leakage Rate (TS 4.4.A)

Proposed TS 4.4.A requires that the containment integrated leakage rate test be performed at a pressure of Pa (47 psig), be conducted over a 24-hour period, and be conducted at 3 approximately equal intervals over each 10-year period. It also specifies that the measured leakage rate not exceed 0.75 La (where La equals 0.1 w/o per day of containment atmosphere at 47 psig).

Evaluation. Proposed TS 4.4.A is in conformance with Section III.A of Appendix J and the procedures of ANSI N45.4-1972. The value of Pa is the peak calculated accident pressure for the IP-2 containment, and the value of La was determined to ensure that public exposure would remain well below the levels of 10CFR100 during a design basis accident. These proposed revisions are therefore acceptable.

3.2.1.2 Sensitive Leakage Rate (TS 4.4.B)

Proposed TS 4.4.B requires that sensitive leakage rate tests be performed with the containment penetrations, weld channels, and certain double-gasketed seals and isolation-valves interspaces at a minimum pressure of 47 psig and that they be conducted at a frequency of at least every other refueling but in no case at intervals greater than 3 years. It also imposed an acceptance criteria of 0.2% of the containment free volume per day.

Evaluation. Proposed TS 4.4.B meets the requirements of Appendix J, Section III.B.1.(c) and III.D.2. The proposed specification is therefore acceptable.

3.2.1.3 Airlock Tests (TS 4.4.C)

Proposed TS 4.4.C requires a containment airlock test to be performed at a minimum pressure of 47 psig (Pa) once every six months with a verification at 47 psig of the double-gasketed airlock-door seals upon closing the airlock door when containment integrity is required.

Evaluation. Proposed TS 4.4.C meets the requirements of Appendix J, Section III.D.2 in that the airlocks are being tested at peak calculated accident pressure (Pa) every six months. However, Section III.D.2 also requires that airlocks be tested at Pa after each use. CEC's proposal to verify the airlock door seals following each use by pressurizing between the double-gasketed seals to a pressure of Pa requires an exemption from the requirements of Section III.D.2.

Experience has shown that for some operating reactors, it is impractical to leak test an airlock at peak calculated accident pressure (Pa), especially when frequent airlock usage is necessary. Testing is a time consuming process and may result is unnecessary exposure to operating personnel. Since the inner door is exposed to pressure in the direction opposite that of the pressure which would exist under accident conditions, strong-backs or other mechanical adjustments are often necessary to prevent the inner door from unseating and preventing meaningful test results. The employment of strong-backs or other mechanical adjustments may even cause a degradation of the airlock operating mechanisms which could eventually lead to reduced airlock reliability.

Since 1969, there have been approximately 40 instances where airlock leak tests have resulted in greater than allowable leak rates. However, they were all caused by the failures of door seals, not by the entire doors. Testing at a pressure of Pa between the double-gasketed seals at IP-2 is an acceptable method to detect door seal leakage while at the same time eliminating the impracticalities, and perhaps reduction of reliability, associated with full airlock testing at Pa.

In view of the above discussion, FRC finds that CEC's proposal to perform a verification of airlock door seals at 47 psig by pressurizing between the double-gasketed seals is an acceptable alternative to performing a Type B test of the airlock after each use and that an exemption from this requirement of Appendix J is acceptable. Consequently, proposed TS 4.4.C is also acceptable.

3.2.1.4 Containment Isolation Valves (TS 4.4.D)

Proposed TS 4.4.D provides for the testing of containment isolation valves, the frequency of the tests, the acceptance criteria, and the addition of valves to the testing list. Each of these provisions are separately evaluated below.

Testing of Containment Isolation Valves

Proposed TS 4.4.D requires that isolation valves be tested in accordance with Table 4.4-1. Table 4.4-1 provides a listing (9 pages) of containment isolation valves, the designated test medium, and the minimum test pressure. The table also indicates which valves are sealed by water systems or by the weld channel and penetration pressurization system. The test pressures are listed as 47 psig (Pa) or 52 psig (110% Pa) in the case of seal water systems.

Evaluation. This portion of TS 4.4.D is in accordance with Appendix J, Section III.C and therefore is acceptable.

Frequency of the Tests

Proposed TS 4.4D requires Type C tests, including the tests of the isolation valve seal water system and the weld channel and penetration pressurization system, be performed at intervals no greater than 2 years.

Evaluation. Section III.D.3 of Appendix J requires that Type C tests be performed during each reactor shutdown for refueling but in no case at intervals greater than 2 years. Since reactor shutdowns for refueling occur, on the average, every 12 to 15 months and since it is impractical to perform these tests in periods other than refueling shotdowns, the basic effect of Section III.D.3 and proposed TS 4.4.D are the same. However, proposed TS 4.4.D is not conservative in relation to the requirements of Section III.D.3 since circumstances can arise in which the frequency of performing Type C tests as required by proposed TS 4.4.D can exceed the frequency between refueling shutdowns as prescribed by Section III.D.3.

Consequently, FRC finds that this portion of proposed TS 4.4.D is not acceptable unless the frequency of the tests is modified to be exactly in accordance with Appendix J, namely that Type C tests be performed during each reactor shutdown for refueling but in no case at intervals greater than 2 years.

Acceptance Criteria

Proposed TS 4.4.D requires that the combined leakage rate of the Type B and Type C tests (other than those Type C tests performed as part of the tests of the seal water systems) be less than 0.6 La. The proposed TS also requires that the leakage rate into the containment for isolation valves sealed by the service water system not exceed 0.36 gpm per fan cooler and that the leakage rate of the isolation valve seal water system not exceed 14,700 cc/hr.

Evaluation. The limit of the combined leakage rate (less than 0.6 La) is in accordance with Section III.C.3 of Appendix J. The limitation of leakage into the containment of 0.36 gpm per fan cooler will permit a full 12 months of post-accident recirculation without flooding the internal recirculation pumps. The limitation of 14,700 cc/hr for the isolation valve seal water system is consistent with the capacity of the system supply tank. The system is designed to provide a supply of 30 days of water post-accident as required by Section III.C.3 of Appendix J. Consequently, FRC finds that this portion of proposed TS 4.4.D is acceptable.

Addition of Valves to the Testing List

Proposed TS 4.4.D permits addition of isolation valves to plant systems without prior licensee amendment provided that a revision to the Table 4.4.-1 is included in a subsequent license amendment application.

Evaluation. Appendix J does not require that the containment leakage testing program be amended prior to any system modifications which effect containment isolation. Appendix J does require that any modifications after to the preoperational containment leak rate testing be followed by appropriate integrated leakage rate or local leakage rate testing. This requirement is imposed by proposed TS 4.4.E below. Consequently, FRC finds that CEC's proposal to include plant modifications in subsequent license amendment applications for revision to Table 4.4-1 to be acceptable.

3.2.1.5 Containment Mondifications (TS 4.4.E)

Proposed TS 4.4.E requires that a containment integrated leakage rate test or a local leak rate test be performed following a modification or replacement of components of the containment after the initial preoperational testing.

Evaluation. Proposed TS 4.4.E complies with Appendix J, Section IV.A and is therefore acceptable.

3.2.1.6 Reporting of Test Results (TS 4.4.F)

Proposed TS 4.4.F requires that integrated containment leakage test results be submitted to the NRC in accordance with Appendix J.

Evaluation. Proposed TS 4.4.F requires reporting requirements as specified in Appendix J and is therefore acceptable.

3.2.1.7 Annual Inspection (TS 4.4.G)

Proposed TS 4.4.G requires that a visual inspection with appropriate corrective action be performed at each refueling shutdown and prior to an integrated leakage rate test in order to uncover evidence of deterioration which might affect either containment structural integrity or leak-tightness with repairs as required.

Evaluation. Proposed TS 4.4.G complies with Appendix J, Section V.A and is therefore acceptable.

3.2.1.8 Residual Heat Removal System (TS 4.4.H)

Proposed TS 4.4.H requires that hydrostatic testing of the RHR system be performed at every refueling shotdown with an acceptance criteria of 2 gph and provisions for corrective action as required.

Evaluation. CEC's basis for this specification is to limit off-site exposure due to liquid leakage to insignificant levels relative to those calculated for the design basis accident. From the standpoint of containment atmosphere leakage, this testing, which is in excess of the testing requirements of Appendix J, will further ensure the operability of this engineered-safety-feature system which provides an additional function of providing the water seals for certain containment isolation valves. Further, experience has shown that liquid limits in the range of 2 gph are indicative of a well-functioning closed system of this size and type and provides additional confidence as to the condition of the system. Proposed TS 4.4.H is therefore acceptable with respect to containment atmospheric leakage requirements of Appendix J.

4.0 CONCLUSIONS

As a result of the foregoing evaluation, the following requests for exclusion from the Type C testing requirements of Appendix J, submitted by CEC in Reference [3], have been found acceptable because the regulation does not require that they be tested. No exemption is required to exclude these valves from the testing program.

- Exclusion from Type C testing for Valves 753H and 753G, component cooling water return line isolation valves to the recirculation pump motors.
- Exclusion from Type C testing for Valves MOV-822A and MOV-822B, component cooling water return line isolation valves from the residual heat exchangers.
- Exclusion from Type C testing for Valves PCV-1111, two valves, one in each penetration supplying air to the weld channel and penetration pressurization system.

Additionally, proposed technical specification 4.4 (Containment Tests) submitted by CEC in Amendment 1 to the Application for Amendment to Facility Operating License DPR-26, dated December, 1977 (Reference [5]) has been found to be acceptable with one acceptable exemption and the need for one modification as described below.

- Proposed TS 4.4.B requires an exemption from the requirements of Appendix J to permit verification of airlock door seals after each use by pressurizing between the double-gasketed seals at a pressure of Pa in lieu of a complete airlock test. FRC has found CEC's proposed exemption request to be acceptable.
- Proposed TS 4.4.D should be modified to require that containment isolation valves requiring Type C tests as well as the isolation valve seal water system and weld channel and penetration pressurization system be tested during each reactor shutdown for refueling but in no case at intervals greater than 2 years as opposed to its current wording which requires testing at intervals no greater than 2 years.

5.0 REFERENCES

[1] NRC Generic Letter from Mr. Karl Goller, Acting Director for Operating Reactors, to Consolidated Edison Company of New York, Inc., (CEC) dated August 7, 1975.

[2] CEC letter from Mr. W. J. Cahill, Jr., Vice President to Mr. Karl Goller, Acting Director for Operating Reactors, dated September 9, 1975.

[3] CEC letter from Mr. C. L. Newman, Vice President to Mr. Karl Goller, Assistant Director, DOR, dated April 14, 1976.

[4] LeBoeuf, Lamb, Leiby, and MacRay letter to Mr. Ben Rusche, Director, NRR, dated April 16, 1976, which forwarded an Application for Amendment to Facility Operating License DPR-26, dated April 14, 1976.

[5] LeBoeuf, Lamb, Leiby, and MacRay letter of December 16, 1977, which forwarded Amendment 1 to the Application for Amendment to Facility Operating License DPR-26, dated December, 1977.

7590-01

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 AND GRANT OF EXEMPTION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 63 to Facility Operating License No. DPR-26, issued to the 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 makes changes to the Technical Specifications related to primary reactor containment leakage testing. In connection with this action, the Commission has granted an exemption which allows the licensee to verify the airlock door seals after each use by pressurizing between the double-gasketed seals instead of performing the test specified in Appendix J to 10 CFR 50, "Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors."

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 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 application for amendment dated April 16, 1976, as revised January 6, 1978 and December 31, 1979, (2) Amendment No. 63 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 Licensing.

Dated at Bethesda, Maryland, this 28th day of August, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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