April 10, 1997

Mr. Stephen E. Quinn
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, NY 10511

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M96369)

Dear Mr. Quinn:

The Commission has issued the enclosed Amendment No. 190 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated August 7, 1996, and supplemented by letter dated March 12, 1997.

The amendment revises the TS to allow the use of 10 CFR Part 50, Appendix J, Option B, "Performance-Based Containment Leak-Test Program." To implement Option B to Appendix J, the amendment revises TSs to eliminate the reference to the prescriptive Appendix J requirements and references instead NRC Regulatory Guide 1.163, September 1995.

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

Sincerely,

ORIGINAL SIGNED BY:

Jefferey F. Harold, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-247 160051

Enclosures: 1. Amendment No. 190 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

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DATED: April 10, 1997

AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-26-INDIAN POINT UNIT 2

Docket File
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 10, 1997

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Jefferey F. Harold

Jefferey F. Harold, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures: 1. Amendment No. 190to DPR-26
2. Safety Evaluation

cc w/encls: See next page

Stephen E. Quinn
Consolidated Edison Company
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Indian Point Nuclear Generating
Station Units 1/2

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 190
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated August 7, 1996, supplemented March 12, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 190 , 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

S. Singh Bajwa, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 10, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 190

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
4.1-1 thru 4.1-4	4.1-1 thru 4.1-4
4.4-1 thru 4.4-10	4.4-1 thru 4.4-8
6-19	6-19

4.0 SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Interval Extension

Unless otherwise noted, each surveillance requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified interval. Excluded from this provision are the following surveillances whose intervals are solely defined by the applicable Technical Specification paragraphs and cannot be extended.

4.4A Integrated Leakage Rate

4.4C Air Lock Tests

4.4D Containment Isolation Valves (those valves without WCCPPS or IVSWS)

4.13 Steam Generator Tube Inservice Inspection.

Basis

Specification 4.0.1 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are specified to be performed at least once each Refueling Interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed once each Refueling Interval. Likewise, it is not the intent that Refueling Interval surveillances be performed during power operation unless it is consistent with safe plant operation. The limitation of Specification 4.0.1 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specifications

a. Calibration, testing and checking of analog channels, and testing of logic channels shall be performed as specified in Table 4.1-1.

b. Sampling and equipment tests shall be conducted as specified in Tables 4.1-2 and 4.1-3, respectively.

c. Performance of any surveillance test outlined in these specifications is not immediately required if the plant condition is the same as the condition into which the plant would be placed by an unsatisfactory result of that test. Such tests will be performed before the plant is removed from the subject condition that has precluded the immediate need to run the test. If the test provisions require that a minimum higher system condition must first be established, the test will be performed promptly upon achieving this minimum condition. The following surveillance tests, however, must be performed without the above exception:

   - Table 4.1-1 Items 3 and 19
   - Table 4.1-2 Items 1, 2, and 10
   - Table 4.1-3 Items 2 and 6

Basis

A surveillance test is intended to identify conditions in a plant that would lead to a degradation of reactor safety. Should a test reveal such a condition, the Technical Specifications require that either immediately, or after a specified period of time, the plant be placed in a condition which mitigates or eliminates the consequences of additional related casualties or accidents. If the plant is already in a condition which satisfies the failure criteria of the test, then plant safety is not compromised and performance of the test yields information that is not necessary to determine safety limits or limiting conditions for operation of the plant. The surveillance test need not be performed, therefore, as long as the plant remains in this condition. However, this surveillance test should be performed prior to removing the plant from the subject condition that has precluded the immediate need to run the test. In the situation in which the test provisions specify that the test must be performed at some minimum system condition, this condition will first be achieved and the test will be performed promptly thereafter prior to proceeding to a higher system condition.

a. <u>CHECK</u>

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm action, and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems, the minimum checking frequency of once per shift when the plant is in operation, is deemed adequate for reactor and steam system instrumentation.

b. <u>CALIBRATION</u>

Calibrations are performed to ensure the presentation and acquisition of accurate information.

The nuclear flux (linear level) channels are calibrated daily against a heat balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of each refueling shutdown.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies of once-per-day for the nuclear flux (linear level) channels, and once each refueling shutdown for the process system channels is considered acceptable.

c. TESTING

The minimum testing frequency for those instrument channels connected to the safety system is based on an average unsafe failure rate of 2.5×10^{-6} failure/hrs. per channel. This is based on operating experience at conventional and nuclear plants. An unsafe failure is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bona fide signal.

For a specified test interval W and an M out of N redundant system with identical and independent channels having a constant failure rate λ, the average availability A is given by:

$$A = \frac{W - Q\left\{\frac{W}{N-M+2}\right\}}{W} = 1 - \frac{N!}{(N-M+2)!\,(M-1)!}(\lambda W)^{N-M+1}$$

where A is defined as the fraction of time during which the system is functional, and Q is the probability of failure of such a system during a time interval W.

For a 2-out-of-3 system A = 0.9999708, assuming a channel failure rate, λ, equal to $2.5 \times 10^{-6}\ hr^{-1}$ and a test interval, W, equal to 2160 hrs.

This average availability of the 2-out-of-3 system is high, hence the test interval of one quarter is acceptable.

Because of their greater degree of redundancy, the 1/3 and 2/4 logic arrays provide an even greater measure of protection and are thereby acceptable for the same testing interval. Those items specified for quarterly testing are associated with process components where other means of verification provide additional assurance that the channel is operable, thereby requiring less frequent testing.

4.4 CONTAINMENT TESTS

Applicability

Applies to containment leakage.

Objective

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

Specifications

A. INTEGRATED LEAKAGE RATE

1. Test

a. A full-pressure integrated leakage rate test shall be performed at intervals specified in Specification 4.4.A.3 at the peak accident pressure (P_a) of 47 psig minimum.

b. The integrated leakage rate test shall be performed in accordance with 10 CFR 50 Appendix J, Option B, as modified by approved exemptions and in accordance with guidelines contained in Regulatory Guide 1.163, dated September 1995.

c. A test duration shall be used in accordance with 10 CFR 50 Appendix J, Option B, as modified by approved exemptions and in accordance with guidelines contained in Regulatory Guide 1.163, dated September 1995.

d. 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.

e. 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 As Found measured leakage rate shall be less than 1.0 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. Prior to entering a mode where containment integrity is required, the As Left leakage rate shall not exceed 0.75 L_a.

3. Frequency

The integrated leakage rate test frequency shall be performed in accordance with 10 CFR 50 Appendix J, Option B as modified by approved exemptions and in accordance with guidelines contained in Regulatory Guide 1.163, dated September 1995.

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 52 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 every Refueling Interval (R#).

C. AIR LOCK TESTS

1. The containment air locks shall be tested at a minimum pressure of 47 psig. The test shall be performed in accordance with 10 CFR 50 Appendix J, Option B, as modified by approved exemptions and in accordance with guidelines contained in Regulatory Guide 1.163, dated September 1995. The acceptance criteria is included in Specification 4.4.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. All isolation valves in Table 4.4-1 shall be tested for operability in accordance with 10 CFR 50 Appendix J, Option B, as modified by approved exemptions and in accordance with guidelines contained in Regulatory Guide 1.163, dated September 1995.

   b. Isolation valves in Table 4.4-1 which are pressurized by the Weld Channel and Containment Penetration Pressurization System are leakage tested as part of the Sensitive Leakage Rate Test included in Specification 4.4.B.

   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 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 Specification 4.4.C.1, portions of the sensitive leakage rate test described in

Specification 4.4.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 Specifications 4.4.A.2, 4.4.B.2, or 4.4.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

A post-outage report shall be prepared presenting results of the previous cycle's Type B and Type C tests, and Type A, Type B, and Type C tests, if performed during that outage. The technical contents of the report are generally described in ANSI/ANS 56.8-1994, and will be available on-site for NRC review. The report shall also show that the applicable performance criteria are met and serves as a record that continuing performance is acceptable.

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 Interval (#) 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 accordance 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 least once every Refueling Interval#.

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 130°F. With these initial conditions, the peak accident pressure and temperature of the steam-air mixture will not exceed the containment design pressure and temperature of 47 psig and 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 weight percent (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 weight percent per day for 24 hours. With this leakage rate and with minimum containment engineered safeguards operating, the public exposure would be well below 10 CFR 100 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 frequency of the periodic integrated leakage rate test is in accordance with 10 CFR 50 Appendix J, Option B as modified by approved exemptions and in accordance with guidelines contained in Regulatory Guide 1.163, dated September 1995. The ability to use Option B is based on the following major considerations.

The first consideration 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 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 10 CFR 50 Appendix J, Option B, as modified by approved exemptions and in accordance with guidelines contained in Regulatory Guide 1.163 dated September 1995. The specified test pressures are ≥ 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 consistent with 10 CFR 50 Appendix J, Option B, as modified by approved exemptions and in accordance with guidelines contained in Regulatory Guide 1.163, dated September 1995.

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 offsite 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, Option B of 10 CFR 50, Regulatory Guide 1.163, "Performance -Based Containment Leak-Test Program," dated September 1995 and NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."

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) UFSAR - Section 5
(2) UFSAR - Section 5.1.6
(3) UFSAR - Section 14.3.6
(4) UFSAR - Section 6.6
(5) UFSAR - Section 6.5

Special Reports

6.9.2 Special reports shall be submitted to the NRC Regional Administrator of the Region I Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. DELETED

b. DELETED

c. Sealed source leakage in excess of limits (Specification 4.15).

d. The complete results of the steam generator tube inservice inspection (Specification 4.13.C.).

e. Radioactive effluents (Specification 3.9).

f. Radiological environmental monitoring (Specification 4.11).

g. Meteorological monitoring instrumentation (Specification 3.15).

h. Inoperable radiation and hydrogen monitoring instrumentation (Specification 3.5) outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

i. Operation of overpressure protection system (Specification 3.1.A.4).

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time intervals at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B, "Performance-Based Requirements," to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall performance and the performance of individual components.

By letters dated August 7, 1996, and March 12, 1997, Consolidated Edison Company of New York (the licensee) requested changes to the Technical Specifications (TS) for Indian Point Nuclear Generating Unit No. 2 (IP2). The proposed changes would permit implementation of 10 CFR Part 50, Appendix J, Option B, and reference Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, which specifies a method acceptable to the NRC for complying with Option B. The March 12, 1997, supplemental letter did not change the initial letter proposed no significant hazards consideration.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage rate assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. 10 CFR Part 50, Appendix J, "Primary Containment Leakage Testing for Water-Cooled Power Reactors," was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Containment Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B, "Performance-Based Requirements," to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, was developed as a method acceptable to the NRC staff for implementing Option B. This RG states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the RG or other implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant TSs. The licensee has referenced RG 1.163 in the proposed IP2 TS.

Regulatory Guide 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TSs which were transmitted to NEI in a letter dated November 2, 1995. These TSs are to serve as a model for licensees to develop plant-specific TS in preparing amendment requests to implement Option B.

In order for a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B, and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's August 7, 1996, and March 12, 1997, letters to the NRC propose TS changes to permit the use of Option B of the revised 10 CFR Part 50, Appendix J. Option B permits a licensee to choose Type A; or Type B and C; or Type A, B, and C; testing to be done on a performance basis. The licensee has elected to perform Type A, B, and C testing on a performance basis. The TS changes refer to RG 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, which specifies methods acceptable to the NRC for complying with Option B. This requires changes to existing TS 4.0.1, 4.4, and 6.9.2.a. Corresponding bases were also modified.

These TS changes replace specific surveillance requirements related to containment leakage rate testing and the corresponding acceptance criteria and test methods with a requirement to perform the testing as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in RG 1.163, dated September 1995. The licensee chose not to include its performance-based testing program in the TS as an administrative program, as was proposed in the November 2, 1995, letter to NEI discussed above. The November 2, 1995, letter provided guidance to licensees but is not an NRC requirement. The staff has reviewed the licensee's proposed TS changes and finds them consistent with the requirements of 10 CFR Part 50, Appendix J, Option B, in that the changes include general reference in the TS to the RG used by the licensee to develop the performance-based leakage-testing program for IP2. The staff has also compared the proposed TS with the model TS in the November 2, 1995, letter to NEI, and finds them to be consistent with the intent of the model TS, with several exceptions, noted below.

3.1 EXCEPTIONS TO THE MODEL TS GUIDANCE

3.1.1 As-Left and As-Found Leakage Rates

The model TS, in the Bases for TS 3.6.1.1.1, state:

> Reviewer's Note: Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.

As an extension of this concept, the licensee is proposing additional words, beyond the model TS, for TS 4.4.A.2, "Acceptance Criteria," to reflect these acceptance criteria for as-left and as-found Type A leakage rates. The staff has reviewed these additional words and finds that they are consistent with RG 1.163 and NEI 94-01, and are therefore acceptable.

3.1.2 Air Lock Leakage Rate Acceptance Criteria

Proposed TS 4.4.C., "Air Lock Tests," deviates from the model TS in that it does not state separate, individual air lock leakage rate testing acceptance criteria. It is, however, the same as the current TS. The proposed TS adds the measured air lock leakage rate to other Type B and C leakage rates and requires that the sum be less than 0.6 L_a, where L_a is the maximum allowable leakage rate for the containment at peak accident pressure, P_a.

This represents no change from the current TS. Further, the provisions of Option B of Appendix J and RG 1.163 do not require separate, individual air lock leakage rate testing acceptance criteria to be placed in the TS. Based on the foregoing, the staff finds the subject TS to be acceptable.

3.1.3 Containment Purge/Vent Valves

It should be noted that the proposed TS set the Type C test interval for containment purge/vent valves to no more than 30 months. Although the model TS guidance provided in the NRC letter to NEI dated November 2, 1995, contains a requirement to perform leakage rate testing of containment purge valves every 6 months, the TS is in brackets, which means that it may or may not be applicable to a specific plant. The licensee's current TS do not contain a requirement for this more frequent leakage rate testing of containment purge/vent valves, which may be compared to the Appendix J, Option A frequency of once per refueling outage. Further, Option B of Appendix J, RG 1.163, and the subordinate guidance documents do not require the testing of these valves more often than once per 30 months. Therefore, the proposed TS sets the test interval for containment purge/vent valves to no more than 30 months, through adherence to Section C.2. of RG 1.163. The staff finds this to be acceptable.

3.2 SUMMARY

In summary, the staff has reviewed the changes to the TS and associated Bases proposed by the licensee and finds that they are in compliance with the requirements of Appendix J, Option B, and are consistent with the guidance of RG 1.163, and finds them to be consistent with the intent of the model TS, with several exceptions reviewed above, and are therefore acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (61 FR 47976). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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