ATTACHMENT A

PROPOSED TECHNICAL SPECIFICATIONS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
AUGUST, 1996

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.

4.1 OPERATIONAL SAFETY REVIEW

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. <u>TESTING</u>

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 Regulatory Guide 1.163.

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

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

In accordance with 10 CFR 50 Appendix J, Option B as modified by approved exemptions and Regulatory Guide 1.163.

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 Regulatory Guide 1.163. 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 Regulatory Guide 1.163.

   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. <u>CONTAINMENT MODIFICATIONS</u>

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. <u>REPORT OF TEST RESULTS</u>

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. <u>VISUAL INSPECTION</u>

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 Regulatory Guide 1.163. 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 reactoroperation.

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 Regulatory Guide 1.163. 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 consistant with 10 CFR 50 Appendix J, Option B, as modified by approved exemptions and Regulatory Guide 1.163.

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" and NEI 94-01 "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

3.10.4 - Shutdown Bank Insertion Limit, Control Bank Insertion Limits and 3.10.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary). (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

c. T.M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

d. NUREG-0800, Standard Review Plan, US Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

e. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary). (Methodology for Specification 3.10.2 Height Dependent Heat Flux Hot Channel Factor.)

6.9.1.10 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

6.9.1.11 The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

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

ATTACHMENT B

SAFETY ASSESSMENT

AND

BASIS FOR NO SIGNIFICANT HAZARDS
CONSIDERATION DETERMINATION

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
AUGUST, 1996

DESCRIPTION OF CHANGES

The Nuclear Regulatory Commission has amended its regulations to provide a performance-based option for leakage rate testing of containments. This performance-based option, Option B may be used as an alternative to the current requirements in Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," 10 CFR Part 50. In order to implement the performance-based leakage rate testing option, the Technical Specifications must be changed to eliminate reference to the present prescriptive Appendix J requirements. Therefore, Con Edison is proposing changes to the Indian Point Unit No. 2 Technical Specifications that would eliminate the existing prescriptive testing requirements for leakage rate testing of the containment and instead reference NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program." These changes would permit the use of the performance-based leakage rate testing, Option B of 10 CFR 50 Appendix J. In addition, there is a minor editorial correction to the mathematical formula for minimum testing frequency in the basis for Technical Specification 4.1 which does not change the formula.

The operation and operability requirements of the containment and containment penetrations are not affected by the proposed Technical Specification changes. Reducing the leakage rate test frequency for Type A Tests from the current three per 10 years to one per 10 years leads to no perceptible increase in risk. The estimated increase in risk is insignificant because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements. Type B and C tests can identify the vast majority of all potential leakage paths. Reducing the frequency of Type B testing of electrical penetrations should be possible with no adverse impact on risk. The vast majority of leakage paths are identified by LLRTs of containment isolation valves (Type C tests). Based upon NUREG-1493 "Performance-Based Containment Leakage Test Program," it has been found that performance-based alternatives to current local leakage testing requirements are feasible without a significant increase in risk. This increase in risk has been reviewed and judged to be acceptable by the NRC as documented by the recent changes to 10 CFR 50 Appendix J.

The proposed changes do not involve any physical modifications to the plant or modification in the methods of plant operation which could cause an accident or event of a different type than previously analyzed. The operational leakage criteria for the containment and the containment penetrations are not affected by the proposed changes. The accident analysis assumptions are not altered by the proposed changes in containment surveillance frequency. Thus, the margin of safety for design basis accidents is unaffected by the proposed changes. Therefore, the proposed changes to the surveillance intervals for the containment and the containment penetrations do not result in an unreviewed safety question or a significant hazards consideration.

Basis for No Significant Hazards Consideration Determination

The proposed changes do not involve a significant hazards consideration since:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   For Indian Point Unit No. 2 the ILRT as-found measured leakage rate acceptance criteria is changed from 0.75 La to 1.0 La. This change is consistent with the revised 10 CFR 50 Appendix J, NEI 94-01, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J". In addition, an as-found leakage rate acceptance criteria of 1.0 La for Type A tests is consistent with the design basis and accident analysis assumptions. The as-left acceptance criteria remains unchanged at 0.75 La in accordance with the NEI guidance. Therefore, prior to entering an operating mode where containment integrity is required the as-left leakage rate will not exceed 0.75 La. The combined leakage rate for containment isolation valves listed in Technical Specification Table 4.4-1 subject to gas or nitrogen pressurization testing, air lock testing, and portions of the sensitive leakage rate test which pertain to containment penetrations and double-gasketed seals shall be less than 0.6 La. The extensive operations and testing experience derived from industry show that risk to the general population is generally insensitive to changes in the allowable leakage rate. It has been determined that the allowable containment leakage can be increased by one to two orders of magnitude without significantly impacting the estimates of population dose in the event of an accident. Furthermore, the Indian Point Unit No. 2 ILRT test history provides substantial justification for the proposed changes. Test results demonstrate that IP-2 has a low leakage containment and that the proposed changes would not jeopardize the ability of the containment to maintain the leakage rate at or below the required limits. The proposed change to Technical Specification 4.1 Basis represent a minor editorial correction to the mathematical formula for minimum testing frequency which does not change the formula. Therefore, the probability and the consequence of a design basis accident are not being increased by the proposed changes.

2 The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Plant systems and components will not be operated in a different manner as a result of the proposed Technical Specification change. The proposed change permits a performance-based approach to determining the leakage-rate test frequency for the containment and containment penetrations (Type A, B, and C tests). There are no plant modifications, or changes in methods of operation. Therefore, the changes in testing intervals for the containment and containment penetrations have no affect on the probability of occurrence of a LOCA. The Limiting Conditions for Operation are not being changed. Changing the as-found leakage-rate acceptance criterion to 1.0 La does not increase the probability or consequences of an accident. Changing the test interval for the containment and containment penetrations does not create any new accident precursors or methods of operation. The proposed change to Technical Specification 4.1 Basis represent a minor editorial correction to the mathematical formula for minimum testing frequency which does not change the formula. Therefore, the possibility for an accident of a different type than was previously evaluated in the safety analysis report is not created by the proposed Technical Specification.

3. The proposed change does not involve a significant reduction in a margin of safety.

While the proposed changes do increase the probability for malfunction of equipment important to safety due to the longer intervals between leakage tests, it has been estimated that the longer test intervals will have an insignificant increase in the overall accident risk to the public. This increase has been reviewed and found to be acceptable by the NRC as documented in NUREG-1493 and the recent rulemaking to 10 CFR 50 Appendix J. We also agree that this increase in accident risk is insignificant. Changing the as-found acceptance criterion to 1.0 La does not increase the consequences of an accident, since the accident analysis assume a leakage rate of La for design basis accidents. The as-left Type A test acceptance criterion remains at less than 0.75 La. Given that the Indian Point Unit No. 2 ILRT test history show no failures during plant life, the proposed changes should not lead to a significant probability of creating new leakage paths or increased leakage rates. The proposed change to Technical Specification 4.1 Basis represent a minor editorial correction to the mathematical formula for minimum testing frequency which does not change the formula. Therefore, the accident analysis assumptions for design basis accidents are unaffected and the margin of safety is not decreased by the proposed Technical Specification change.

Based on the preceding analysis it is concluded that operation of Indian Point Unit No. 2 in accordance with the proposed amendment does not increase the probability of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, nor reduces any margin of plant safety. Therefore, the license amendment does not involve a Significant Hazards Consideration as defined in 10 CFR 50.92.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Nuclear Facilities Safety Committee (NFSC). Both Committees concur that the proposed changes do not represent a significant hazards consideration.