ATTACHMENT I
PROPOSED TECHNICAL SPECIFICATION CHANGES

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
SEPTEMBER, 1994

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

2. Acceptance Criteria

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

3. Frequency

In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

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.

ATTACHMENT II
SAFETY ASSESSMENT

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
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DESCRIPTION OF PROPOSED CHANGES

"Type A Tests" are defined in Appendix J, Section II.F as "tests intended to measure the primary reactor containment overall integrated leakage rate."

Exemption is requested from the following portion of Appendix J, Section III.D.1.(a) for Type A test intervals:

> "Type A tests shall be performed, 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 inservice inspections."

The proposed exemption to 10 CFR 50, Appendix J, Section III.D.1.(a), "Type A Periodic Retest Schedule," would allow for a one-time extension of the interval between the second and third Type A test during the second ten year service period. The extension would allow the Type A integrated leak rate test (ILRT) to be performed at the thirteenth refueling outage instead of the twelfth refueling outage as currently scheduled.

The purpose of Appendix J leak test requirements, as stated in the Introduction to 10 CFR 50 Appendix J, is to "assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the technical specifications or associated bases and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment."

This exemption request concerns part (a) of the stated purpose of Appendix J. Part (b) of the stated purpose of Appendix J applies to penetrations and isolation valves, which are tested by Type B and C Local Leak Rate Tests (LLRTs).

The proposed change to the IP 2 Technical Specification would revise (TS) 4.4.A.3 such that it would reference 10 CFR 50, Appendix J directly, rather than paraphrase the regulation, and allow approved exemptions to the ILRT frequency requirements.

REGULATORY BASIS FOR SPECIFIC EXEMPTION

Pursuant to 10 CFR 50.12(a)(2), the NRC will not consider granting an exemption to a requirement unless special circumstances are present. This exemption request meets the special circumstances of paragraphs (a)(2)(ii) and (a)(2)(vi) of 10 CFR 50.12. The exemption request, as discussed below, demonstrates that: the underlying purpose of the regulation is achieved [(a)(2)(ii)] and there are present material circumstances not considered when the regulation was adopted [(a)(2)(vi)]. The granting of this requested exemption will not present an undue risk to the health and safety of the public and is consistent with the common defense and security.

BACKGROUND INFORMATION

The NRC is currently examining those regulations which may be revised to reduce regulatory burden on licensees without a significant impact on safety. As part of this effort, the NRC is currently processing a proposed revision to 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The current proposal for a revised Appendix J will relax the schedule requirements for Type A Integrated Leak Rate Tests (ILRTs) and change the schedule for Type B and C Local Leak Rate Tests (LLRTs) to a performance-based schedule.

According to SECY 94-036, this proposed rule change will not be approved until approximately August, 1995. Therefore, licensees who have refueling outages scheduled prior to August 1995 will not be able to implement the revised rule to make use of the relaxed requirements during these refueling outages.

The Cycle 12 refueling outage for Indian Point 2 is currently scheduled to begin in February 1995. Consolidated Edison (Edison) is therefore requesting a one-time exemption to the Type A ILRT test schedule for the Cycle 12 refueling outage. Although changes to the acceptance criteria for leak rate tests and schedules for Type B and C tests are part of the proposed rule change, an exemption and change is <u>not</u> currently being sought for these requirements.

DESCRIPTION OF CONTAINMENT

The IP2 Containment is a reinforced concrete shell in the form of a vertical right cylinder with a hemispherical dome and a generally flat base supported on rock. The inside surface of the structural concrete is lined with 1/4 inch minimum thickness steel plate anchored to the concrete shell. The liner is designed and fabricated to prevent leakage through it if an accident occurs resulting in the loss of reactor coolant and release of radioactive material to the containment volume concurrent with an earthquake.

The containment has side walls which are 147 feet high from the liner on the horizontal base to the spring line of the dome and has a 135 foot inside diameter. The containment free volume is 2,610,000 cubic feet. The thickness of the reinforced concrete base is 9 feet, the side walls are 4.5 feet, and the dome is 3.5 feet thick. The bottom horizontal liner plate is covered with two feet of concrete, the top of which forms the floor of containment.

The liner is anchored to the concrete shell by means of Nelson studs so that it becomes part of the entire structure under all loadings in such a manner as to insure leak tightness.

All penetrations made in the structure have been considered as potential leak sources and as such are designed with double barriers and treated by a leak prevention system. The weld seams required to fabricate the steel liner are also considered as potential leak sources and are barriered and treated by the leak prevention system (Weld Channel and Containment Penetration Pressurization System). Further information on the Containment design can be found in section 5.1 of the IP 2 Updated Final Safety Analysis Report (UFSAR).

Historical Type A Testing Results

10 CFR 50, Appendix J, Section II.K defines the acceptable leakage limit L_a as, "the maximum allowable leakage rate at pressure P_a [calculated design basis accident peak containment pressure] as specified for preoperational tests in the technical specifications or associated bases, and as specified for periodic tests in the operating license."

The IP 2 Type A test history provides substantial justification for the proposed test schedule. As can be seen below, five (5) Type A tests have been performed to date and considerable margin exists between the Type A test results and the Technical Specification 4.4.A.2 limit of 0.75% L_a where L_a is equal to 0.1% per day of containment atmosphere at a peak accident pressure of 47 psig (61.696 psia). These tests demonstrate that IP 2 has a low leakage containment and that the proposed 24 month extension would not jeopardize the ability of the containment to maintain the leakage rate at or below the required Type A limits.

Two different testing methods were employed in performing some of the tests; the mass point leakage rate method and the total time leakage rate method. When more than one method was utilized, the results of both tests are reported. The results of the individual IP 2 Type A tests follow:

First Periodic Type A Test

The first periodic Type A test was successfully completed on July 27, 1976 with the following results: 1) A calculated mass point leakage rate of 0.0% per day, and 2) A total time leakage rate of 0.0% per day. The test was performed at a pressure of 65.362 psia.

Second Periodic Type A Test

The second periodic Type A test was successfully completed on August 20, 1979 with the following results: 1) A calculated mass point leakage rate of 0.025% per day, and 2) a 95% upper confidence limit (UCL) of .026% per day. The test was performed at a pressure of 61.7 psia.

Third Periodic Type A Test

The third periodic Type A test was successfully completed on September 20, 1984 with the following results: 1) A calculated as found mass point leakage rate of 0.031% per day and an as found 95% UCL of .032% per day, 2) A calculated as left mass point leakage rate of 0.027% per day and an as left 95% UCL of .028% per day. The test was performed at a pressure of 65.5 psia. This test corresponded to the end of the first ILRT 10-year testing period.

Fourth Periodic Type A Test

The fourth periodic Type A test was successfully completed on December 22, 1987 with the following results: 1) A calculated mass point leakage rate of 0.046279% per day and a 95% UCL of .047726% per day, 2) A total time leakage rate of .032925% per day and a 95% UCL of .033752% per day. The test was performed at a pressure of 61.696 psia.

Fifth Periodic Type A Test

The fifth periodic Type A test was successfully completed on June 22, 1991 with the following results: 1) A calculated mass point leakage rate of 0.045934 per day and a 95% UCL of .047791% per day. The test was performed at a pressure of 61.696 psia.

DISCUSSION OF CHANGE

Factors affecting leak tightness of containment may be categorized as: 1) active components which are leak rate tested by Type B and C tests, and 2) passive components which constitute the containment structure and are tested during the Type A test.

Active Components

The purpose of containment leak testing is to detect any containment leakage resulting from active or passive failures in the containment isolation boundaries before an accident occurs. The major containment leakage paths include:

1) **Penetration Seal Leakage:** Air lock door seals; doors with resilient seals or gaskets except for seal welded doors; and penetrations whose design incorporates resilient seals, gaskets, or sealant compounds, piping penetrations fitted with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies may all exhibit leakage. Type B tests cover this type of leakage and will not be affected by the proposed change in the ILRT test schedule.

2) **Containment Isolation valves:** These valves provide either a potential or direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, are required to close automatically upon receipt of a containment isolation signal in response to controls intended to affect containment isolation and are required to operate intermittently under post accident conditions. Leakage through these valves can be caused by leaking valve seals, isolation valve closure failure, or failure to return a penetration to its normally closed condition following maintenance. For all of these initiating events, except post-maintenance/LLRT errors, this type of leakage is detectable by Type C local leak rate testing. Following any maintenance on a Containment Isolation valve, an LLRT is performed followed by an independent valve alignment verification to ensure that leakage remains within acceptable levels. Type C tests will not be affected by the proposed change in the ILRT test schedule.

3) **Gross Containment failure:** This is a low probability event which is the only event likely to be detected only by a Type A test.

The existing Type B and C testing programs are not being modified by this request and will continue to effectively detect containment leakage caused by the degradation of active containment isolation components (e.g., valves) as well as sealing material within containment penetrations.

Industry experience indicates that 97% of the failures associated with Type A tests are found to be due to Type B and C tested penetrations (Draft NUREG 1493, "Performance-Based Containment Leak Test Program"). The local leak rate testing frequencies of these penetrations are not affected by this proposed change. Therefore, continued overall leak tightness of the active containment components can be assured by the existing Type B and C testing program.

Passive Structure

Two mechanisms could adversely affect the passive structural capability of containment. The first is deterioration of the structure due to pressure, temperature, radiation, chemical, or other such effects. Secondly, modifications can be made to the structure which, if not carefully controlled, could leave the structure with reduced capability.

Absent actual accident conditions, structural deterioration is a gradual phenomenon requiring periods of time well in excess of the proposed interval extension. Other than accident conditions, the only pressure challenge to containment is the ILRT itself.

10 CFR 50, Appendix J, Section V.A requires a general inspection of accessible interior and exterior surfaces of the containment structures and components to be performed prior to any Type A test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak tightness. At IP 2 there has been no evidence of structural deterioration that would impact structural integrity or leak tightness.

Modifications that would alter the passive containment structure are infrequent and would receive extensive review to ensure containment capabilities are not diminished. The IP 2 design change and 10 CFR 50.59 programs have been demonstrated to be effective in providing a high quality oversight of such safety significant modifications. In addition, 10 CFR 50, Appendix J, Section IV.A, requires Type A testing to be performed following any major modification to the primary containment boundary. This requirement will be maintained.

Risk Impact Assessment

The risk impact of containment structural life is measured by a pathway created for radionuclides if the containment is challenged such as in a loss of coolant accident (LOCA) or severe accident. Such leakage does not create any new accident scenarios, nor does it contribute to the initiation of any accident.

From a risk standpoint, the purpose of Appendix J leak testing is to detect any containment leakage resulting from failures in the containment isolation boundary before an accident occurs. Such leakage could be the result of leakage through containment penetrations, through airlocks, or through containment structural faults. The Appendix J Type B and C tests, which are unaffected by this proposed change, will continue to detect leakage through containment valves, penetrations, and airlocks. The only potential failures that would not be detected by Type B and C testing are mechanical failures of the containment shell (i.e., degradations or modifications to the containment shell). Thus, the only potential effect of the proposed one-time change to the Type A test frequency is the probability that containment structural leakage would go undetected between tests.

The containment structure is passive. Under normal operating conditions, there is no significant environmental or operational stress present that could contribute to its degradation. A review of modifications for potential effects to the containment structure is described in the preceding section. Passive failures resulting in significant containment structural leakage are therefore extremely unlikely to develop between Type A tests. No such failures have occurred at IP2.

Postulated containment failure under severe accident conditions is primarily due to phenomenological effects associated with severe accidents. Such effects were considered as part of the IP2 IPE. None of the identified containment failure mechanisms for severe accidents would be impacted by the proposed increase in the testing interval.

Based on information provided in Draft NUREG-1493, the increased risk of population dose attributable to extending the test interval from three to five years would be extremely small.

Draft NUREG-1493 includes the results of a sensitivity study performed to explore the risk impact of several alternate leak rate testing schedules. "Alternative 4" from this study examines relaxing the ILRT frequency from 3 in 10 years to 1 in 10 years. Using best estimate data, the draft NUREG concludes that the increase in population exposure risk to those in the vicinity of the five representative plants ranged from 0.02 to 0.14%. This very low impact on risk is attributable to 1) the effectiveness of Type B and C tests in identifying potential leak paths (about 97%), 2) a low likelihood of ILRT-identified leakages in excess of 2 times allowable, and 3) the insensitivity of risk to containment leak rate, (e.g. no discernible increase in population dose risk with containment leak rates 100 times greater than currently allowed). This led the authors of draft NUREG-1493 to conclude that even increasing the ILRT frequency to once per 20 years would "lead to an imperceptible increase in risk."

The exemption requested for IP 2 is concluded to be bounded by the analyses of draft NUREG-1493 because the requested exemption would result in a one-time test interval of five years; not 20 or even 10 years. Consolidated Edison believes that there is sufficient information in the Draft NUREG-1493 to conclude that the risk increase from the requested exemption is low and that the value, in terms of enhanced public safety, of performing the ILRT in 1995 is extremely low.

BASIS FOR EXEMPTION

The proposed interval extension meets the criteria for special circumstances as described in 10 CFR 50.12(a)(2)(ii) and (vi).

50.12(a)(2)(ii) Application of the Regulation is not Necessary to Achieve the Underlying Purpose of the Rule

The underlying purpose of 10 CFR 50, Appendix J is still achieved. Appendix J states that the leakage test requirements set forth in this appendix provide for periodic verification by tests of the leak tight integrity of the primary reactor containment. The appendix further states that the purpose of the tests is to assure that leakage through the primary reactor containment shall not exceed the allowable leakage rate values as specified in the technical specifications or associated bases.

10 CFR 50, Appendix J, Section III.D.1.(a) states that a set of three periodic tests shall be performed at approximately equal intervals during each 10-year period and that the third test shall be conducted when the plant is shutdown for the 10-year plant inservice inspections. This exemption would eliminate the third Type A test from the second ten year interval. The methodology, acceptance criteria, and technical specification leakage limits for the performance of the Type A test will not change.

The testing history, structural capability of the containment, and the risk assessment discussed previously establish that IP2 has had acceptable containment leakage rates, that the structural integrity of containment is assured, and that there is negligible risk impact in changing the Type A test schedule on a one-time basis.

This exemption request does not affect the periodic schedule for Type B and C tests which will continue to be performed in accordance with Appendix J and approved exemptions. Demonstrated operability of the associated components and penetrations through Type B and C tests adds assurance that the overall Type A leakage rates remain satisfactory. No significant leakage trends have been identified which threaten the overall containment leakage specifications. There is no significant change in the types or increase in the amounts of any effluents that may be released offsite due to the elimination of the performance of the third Type A test during the second ten year interval. This one-time change does not impact the design basis of the plant and would not affect the response of containment during a design basis accident.

Thus, there is significant assurance that the extended interval between Type A tests will continue to provide periodic verification of the leak tight integrity of the containment.

50.12(a)(2)(vi) Presence of Material Circumstances not Considered when the Regulation was Adopted

Certain material circumstances were not considered when the regulation was adopted. The benefit of time has produced experience and information that provide a better perspective about containment integrity. Two important material circumstances are testing history and the development of probabilistic risk assessments (PRAs).

Since the promulgation of 10 CFR 50, Appendix J, in 1973, more than 20 years of nuclear power plant operating history has been obtained. A review of industry data did not find any instances where a Type A test failed to meet Appendix J acceptance criteria as a result of a containment structural leak not due to initial fabrication or a plant modification. This additional operating history provides a significant indicator that containment structural integrity (passive) is not a significant safety concern.

Plant specific PRAs were not available and therefore were not considered when the regulation requiring compliance with Appendix J (10 CFR 50.54(o)) was adopted. Overall plant risk due to containment leakage is relatively small given the small probability of containment leakage itself. The predominant contributor to degraded containment integrity is the phenomenological effects of a severe accident, not preexisting containment isolation conditions.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

An evaluation of the proposed exemption to 10 CFR 50, Appendix J requirements and the proposed administrative Technical Specification change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this exemption and amendment request follows:

Criterion I - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed change will provide a one-time exemption from the 10 CFR 50, Appendix J Section III.D.I.(a) leak rate test schedule requirement. This change will allow for a one-time test interval for Type A Integrated Leak Rate Tests (ILRTs) of approximately 70 months.

Leak rate testing is not an initiating event in any accident, therefore this proposed change does not involve a significant increase in the probability of a previously evaluated accident.

Type A tests are capable of detecting both local leak paths and gross containment failure paths. The history at IP 2 demonstrates that Type B and C Local Leak Rate Tests (LLRTS) have consistently detected any excessive local leakages.

Administrative controls govern the maintenance and testing of containment penetrations such that the probability of excessive penetration leakage due to improper maintenance or valve misalignment is very low. Following maintenance on any containment penetration, an LLRT is performed to ensure acceptable leakage levels, Following any LLRT on a containment isolation valve, an independent valve alignment check is performed. Therefore, Type A testing is not necessary to ensure acceptable leakage rates through containment penetrations.

While Type A testing is not necessary to ensure acceptable leakage rates through containment penetrations, Type A testing is necessary to demonstrate that there are no gross containment failures. Structural failure of the containment is considered to be a very unlikely event, and in fact, since IP 2 has been in operation it has never failed a Type A ILRT. Therefore, a one-time exemption increasing the interval for performing an ILRT should not result in a significant decrease in the confidence in the leak tightness of the containment structure.

The proposed change also revises Technical Specification 4.4.A.3 to reference the testing frequency requirements of 10 CFR 50, Appendix J, and to state that NRC approved exemptions to the applicable regulatory requirements are permitted. The current language of TS 4.4.A.3 paraphrases the requirements of Section III,D.I.(a) of Appendix J. The proposed administrative revision simply deletes the paraphrased language and directly references Appendix J. No new requirements are added, nor are any existing requirements deleted. Any specific changes to the requirements of Section III.D.I.(a) will require a submittal from Consolidated Edison under 10 CFR 50.12 and subsequent review and approval by the NRC prior to implementation. The proposed change is stated generically to avoid the need for further TS changes if different exemptions are approved in the future.

The proposed change, in itself, does not affect reactor operations or accident analysis and has no radiological consequences. The change provides clarification so that future Technical Specifications changes will not be necessary to correspond to applicable NRC approved exemptions from the requirements of Appendix J.

Therefore, this proposed change does <u>not</u> involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed exemption request does not affect normal plant operations or configuration, nor does it affect leak rate test methods. The proposed change allows a one-time test interval of approximately 70 months for the ILRT. Given the test history of IP 2 of no Type A test failures during plant lifetime, the relaxation in schedule should not significantly decrease the confidence in the leak tightness of the containment.

The proposed Technical Specification amendment provides clarification to a specification that paraphrases a codified requirement.

Since the proposed change would not change the design, configuration or method of operation of the plant, it would not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

The purpose of the existing schedule for ILRTs is to ensure that the release of radioactive materials will be restricted to those leak paths and leak rates assumed in accident analyses. The relaxed schedule for ILRTs does not allow for relaxation of Type B and C LLRTs. Therefore, methods for detecting local containment leak paths and leak rates are unaffected by this proposed change. Given that the test history for ILRTs shows no failure during plant life, a one-time increase of the test interval does not lead to a significant probability of creating a new leakage path or increased leakage rates, and the margin of safety inherent in existing accident analyses is maintained.

The proposed Technical Specification change is administrative and clarifies the relationship between the requirements of TS 4.4.A.3, Appendix J and any approved exemptions to Appendix J. It does not, in itself, change a safety limit, an LCO, or a surveillance requirement on equipment required to operate the plant. The NRC will directly approve any proposed change or exemption to III..D.1.(a) of Appendix J prior to implementation.

Therefore, this change does not involve a significant reduction in the margin of safety.

Based on the Safety Analysis, it is concluded that: (1) The proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92 and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change. Moreover, because this action does not involve a significant hazards consideration, it will also not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

The proposed changes have been reviewed by the Station Nuclear Safety Committee which concurs that the requested change does not involve a significant hazards consideration. The changes will be reviewed by the Nuclear Facilities Safety Committee prior to implementation.