ATTACHMENT A

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Technical Specification
Page Revisions

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
March, 1981

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1.8 Quadrant Power Tilt Ratio

The quadrant power tilt ratio shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average. (W-STS)

1.9 Surveillance Intervals

Unless otherwise noted in an individual surveillance requirement, surveillance intervals shall be as specified in Table 1-1 with extensions as provided in 1.10 below. The extensions provided in 1.10 below also apply to surveillance intervals not listed in Table 1-1 unless the extensions are specifically not allowed.

1.10 Surveillance Interval Maximums

Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. A total maximum combined interval time for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval. (W-STS)

1.11 Pressure Boundary Leakage

PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolatable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

1.12 IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be:

a. Reactor coolant system leakage into closed systems such as pump seal or valve packing leaks that are captured and conducted to a collecting tank, or

b. Reactor coolant system leakage through a steam generator to the secondary system, or

c. Reactor coolant system leakage through the RCS/RHR pressure isolation valves, or

d. Reactor coolant system leakage into the containment free volume from sources that are both specifically located and known either not to interfere with the operation of required leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

1.13 UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all reactor coolant system leakage which is not IDENTIFIED LEAKAGE.

3.1.F. REACTOR COOLANT SYSTEM LEAKAGE AND LEAKAGE INTO THE CONTAINMENT FREE VOLUME

Specification

1. LEAKAGE DETECTION AND REMOVAL SYSTEMS

a. The reactor shall not be brought above cold shutdown unless the following leakage detection and removal systems are operable:

(1) Two containment sump pumps.
(2) Two containment sump level monitors.
(3) A containment sump discharge line flow monitoring system.
(4) Two recirculation sump level monitors.
(5) The reactor cavity continuous level monitoring system and an independent reactor cavity level alarm.
(6) Two of the following three systems:
(a) A containment atmosphere gaseous radioactivity monitoring system.
(b) A containment atmosphere particulate radioactivity monitoring system.
(c) The containment fan cooler condensate flow monitoring system.

b. When the reactor is above cold shutdown, the requirements of specification 3.1.F.1.a may be modified as follows:

(1) One containment sump pump may be inoperable for a period not to exceed seven (7) consecutive days provided that on a daily basis the other containment sump pump is started and discharge flow is verified.

(2) One of the two required containment sump level monitors may be inoperable for a period not to exceed seven (7) consecutive days.

(3) The containment sump discharge line flow monitoring system may be inoperable for a period not to exceed seven (7) consecutive days provided a detailed Waste Holdup Tank water inventory balance is performed daily.

(4) One of the two required recirculation sump level monitors may be inoperable for a period not to exceed fourteen (14) consecutive days.

(5) Either the reactor cavity continuous level monitoring system or the required independent reactor cavity level alarm may be inoperable for a period not to exceed thirty (30) consecutive days.

(6) Two of the three monitoring systems specified in specification 3.1.F.1.a.(6) may be inoperable for a period not to exceed thirty (30) consecutive days. If both radioactivity monitoring systems specified in specification 3.1.F.1.a.(6) are inoperable, operation may continue for a period not to exceed thirty (30) days provided grab samples of the containment atmosphere are obtained and analyzed daily.

c. If the conditions of specification 3.1.F.1.b cannot be met or an inoperable system(s) is not restored to operable status within the time period(s) specified therein, then, either perform a visual inspection of containment at least once a shift, or place the reactor in the hot shutdown condition within the next 6 hours and, if the inoperability continues, place the reactor in the cold shutdown condition within the following 30 hours.

2. OPERATIONAL LEAKAGE LIMITS

a. Primary to Secondary Leakage:

(1) Primary to secondary leakage through the steam generator tubes shall not exceed 0.3 gpm in any steam generator. With any steam generator tube leakage greater than this limit, the reactor shall be brought to the cold shutdown condition within 24 hours.

(2) If leakage from two or more steam generators in any 20-day period is observed or determined, the reactor shall be brought to the cold shutdown condition within 24 hours and Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation. If two steam generator tube leaks attributable to the tube denting phenomena are observed after the reactor is in cold shutdown, Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

(3) Whenever the reactor is shutdown in order to investigate steam generator tube leakage and/or to plug or otherwise repair a leaking tube, the NRC shall be informed before any tube is plugged or, if no tube is plugged, before the steam generator is returned to service.

b. RCS/RHR Pressure Isolation Valves Leakage:

(1) Whenever the reactor is above cold shutdown, leakage through each of the RCS/RHR pressure isolation valves 879A, B, C & D and 838A, B, C & D shall satisfy the following acceptance criteria:

(a) Leakage rates less than or equal to 1.0 gpm are acceptable.

(b) Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between the measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

(c) Leakage rates greater than 5.0 gpm are unacceptable.

(2) If any RCS/RHR pressure isolation valve listed in specification 3.1.F.2.b.(1) is determined to be inoperable based on the acceptance criteria presented therein, an orderly plant shutdown shall be initiated and the reactor shall be placed in the cold shutdown condition within 24 hours.

c. Total Reactor Coolant System Leakage:

(1) Whenever the reactor is above cold shutdown, reactor coolant system leakage shall be limited to:

(a) No PRESSURE BOUNDARY LEAKAGE,

(b) 1 gpm UNIDENTIFIED LEAKAGE, and

(c) 10 gpm IDENTIFIED LEAKAGE.

(2) With any PRESSURE BOUNDARY LEAKAGE, the reactor must be placed in hot shutdown with 6 hours and in cold shutdown within the following 30 hours.

(3) If the Reactor Coolant System leakage exceeds the limits in either c.(1)(b) or c.(1)(c) above, the leakage rate must be reduced to within limits within 4 hours or the reactor must be placed in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

d. Leakage Into The Containment Free Volume:

(1) Whenever the reactor is above cold shutdown, the total leakage into the containment free volume from both reactor coolant and non-reactor coolant sources combined shall not exceed 10 gpm.

(2) Notwithstanding the action which may be required by specification 3.1.F.2.d.(3) below, with the combined leakage into the containment free volume greater than the above limit, the leakage rate must be reduced to within the specified limit within 12 hours or the reactor must be placed in cold shutdown within the following 36 hours.

(3) If water level in the containment sump reaches EL. 45' or the water level in the recirculation sump reaches EL. 34'-11½", or the water level in the reactor cavity reaches EL. 20', the reactor shall be placed in a cold shutdown condition within the next 36 hours unless the water level(s) is reduced below the specified limit(s).

(4) If the water level in the containment sump increases above EL. 46' and the water level in the recirculation sump increases above EL. 39'-9", or the water level in the reactor cavity increases EL. 20'-5", immediately place the reactor in a subcritical condition and initiate an expeditious cooldown of the reactor to the cold shutdown condition.

Basis

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of gpm may be tolerable from a dose point of review, especially if they are to closed systems, it must be recognized that leaks in the order of drops per minute through any pressure boundary of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture.

If leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor is sensitive to low rates. The rates of reactor coolant leakage to which the instrument is sensitive are 0.025 gpm to greater than 10 gpm, assuming corrosion product activity and no fuel cladding leakage. Under these conditions, an increase in reactor coolant system leakage of 1 gpm is detectable within 1 minute after it occurs.

b. The containment radiogas monitor is less sensitive than the air particulate monitor. The sensitivity range of the instrument is $10^{-3}\mu c/cc$ to $10^{-6}\mu c/cc$.

c. A leakage detection system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main air recirculation units including leaks from the cooling coils themselves. This system provides a dependable and accurate means of measuring the total leakage from these sources. Condensate flows from approximately 0.05 gpm to 50 gpm per detector can be measured by this system. Leaks less than 1 gpm may be determined by periodic observation of the water accumulation in the standpipes of the condensate collection system.

d. Leakage detection via the containment sump level and discharge flow monitoring systems will determine leakage losses from all fluid systems to the containment free volume. Water collecting on the containment floor will normally be delivered to the containment sump via the containment floor trench system. Level monitoring of the containment sump is in part provided by two level switch assemblies which actuate control room lights at discrete sump/containment water levels and provide an audible alarm for certain discrete levels within the containment sump. In addition, another level transmitter provides a continuous level readout in the control room. When the water level in the containment sump reaches a predetermined level, one or both containment sump pumps will automatically start and pump the fluid out of containment to the liquid waste disposal system. Flow in the containment sump pump discharge line from containment to the Waste Holdup Tank is monitored on a continuous basis. Thus, monitoring of both the containment sump inventory and discharge via level and flow indication systems will provide a positive means for determining leakage into the containment free volume.

e. Water may also collect in the recirculation sump and/or the reactor cavity depending on the size and location of the leak. However, under most circumstances, the containment sump will be filled prior to the recirculation sump filling and both sumps will be filled prior to water level increasing on containment floor (EL. 46') sufficient to initiate filling of the reactor cavity. Level monitoring of the recirculation sump is provided by two level switch assemblies which actuate control room ligths at discrete sump/containment water levels and provide an audible alarm for certain discrete levels within the recirculation sump. In addition, another level transmitter provides a continuous level readout in the control room. Level monitoring of the reactor cavity is provided by a level transmitter which provides a continuous level readout in the control room and two float switches each of which actuates an audible alarm in the control room.

Total reactor coolant leakage can be determined by means of periodic water inventory balances. If leakage is into another closed system, it will be detected by the plant radiation monitors and/or inventory balances. Determined leakage rates are an average over the applicable surveillance interval. Industry experience has shown that while a limited amount of leakage is expected from the RCS, the UNIDENTIFIED portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure detection of additional leakage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in cold shutdown. Primary system leakage through packing, gaskets, seal welds or mechanical joints is not considered to be PRESSURE BOUNDARY LEAKAGE.

The leakage limit and surveillance testing for RCS/RHR Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS/RHR Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those limits found to result in negligible corrosion of the steam generator tubes. If stress corrosion cracking occurs, the extent of cracking during plant operation would be limited by limitation of steam generator leakage between the reactor coolant system and the secondary coolant system. Leakage in excess of 0.3 gpm for any steam generator will require plant shutdown and the leaking tube(s) will be located and plugged.

The 10 gpm limit for combined reactor coolant and non-reactor coolant leakage into the containment free volume provides allowance for a limited amount of leakage from sources other than the reactor coolant system within containment while conservatively limiting total leakage into the containment free volume to the same limit (i.e., 10 gpm) for identified reactor coolant leakage alone. This leakage is within the capabilities of the leakage detection and waste processing system and will not interfere with the detection of independent unidentified reactor coolant system leakage.

For those circumstances where high energy line failures occur inside containment resulting in flooding of the containment building sumps and/or floor, automatic actuation of reactor protection, safety injection and/or containment spray systems places the plant in a safe condition and, in some cases, provides intended flooding of the containment building. However, for those circumstances resulting from leakage or failure of low energy systems such as service water or component cooling inside containment, operator action is necessary to prevent accumulation of water on the containment floor to undesirable levels.

If the water level in the containment sump reaches EL. 45' or the water level in the recirculation sump reaches EL. 34-11½" or the water level in the reactor cavity reaches EL. 20' an orderly shutdown is initiated. If the water level in the containment sump increases above EL. 45' and the water level in the recirculation sump increases above EL. 39'-9", or the water level in the reactor cavity increases above EL. 20'-5", the operator will immediately bring the reactor subcritical and initiate an expeditious cooldown of the plant.

The above actions are necessary to: (1) preclude accumulation of water inside containment such that if a LOCA were to occur safety-related equipment would not become submerged, (2) prevent the reactor cavity from becoming filled with water, (3) prevent the reactor vessel from being wetted while it is at an elevated temperature, and (4) prevent the immersion of the in-core instrument conduits. The amount of water estimated to be inside containment after actuation of the emergency core cooling system following a loss of coolant accident is approximately 423,000 gallons. This amount of water would, by itself, reach approximately EL. 50'-1". An additional 28,000 gallons (a total of approximately 451,000 gallons) would have to accumulate inside containment before any safety-related electrical component would be submerged (approximately EL. 50'-5"). The combined volume of the containment sump, the recirculation sump and the containment floor trenches is approximately 18,000 gallons. Since operator action is required by these specifications to shut the reactor down before these volumes are filled, sufficient margin between the water level inside containment following a loss of coolant accident and the level at which a safety-related electrical component may become submerged is maintained. Furthermore, since both sumps, the floor trenches and the containment floor up to EL. 46'-5 3/8" must be flooded (i.e., approximately 50,000 gallons) prior to the water level being sufficiently high to flood over the curb leading to the reactor cavity, the forementioned operator actions taken to preclude excessive flooding plus LOCA water levels will conservatively preclude flooding of the reactor cavity and subsequent wetting of the reactor vessel at an elevated temperature.

References
FSAR Sections 6.7, 11.2.3 and 14.2.4

TABLE 4.1-1 (CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S	N.A.	N.A.	With analog rod position
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A	
13.	Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14.	Boric Acid Tank Level	W	R	N.A.	Bubbler tube rodded during calibration
15.	Refueling Water Storage Tank Level	W	R	N.A.	
16.	Boron Injection Tank Level	W	R	R	
17.	Volume Control Tank Level	N.A.	R	N.A.	
18.	(a) Containment Pressure	D	R	M	Wide range
	(b) Containment Pressure	S	R	M	Narrow range
19.	Process and Area Radiation Monitoring Systems	D	R	M	
20.	Boric Acid Make-up Flow Channel	N.A.	R	N.A.	
21A.	Containment Sump and Recirculation Sump Level (Discrete)	S	R	R	Discrete Level Indication Systems.
21B.	Containment Sump, Recirculation Sump and Reactor Cavity Level (Continuous)	S	R	R	Continuous Level Indication Systems.
21C.	Reactor Cavity Level Alarm	N.A.	R	R	Level Alarm System.
21D.	Containment Sump Discharge Flow	S	R	M	Flow Monitor.
21E.	Containment Fan Cooler Condensate Flow	S	R	M	Monthly visual inspection of condensate weirs.

TABLE 4.1-1 (CONTINUED)

Channel Description	Check	Calibrate	Test	Remarks
22. Accumulator Level and Pressure	S	R	N.A.	
23. Steam Line Pressure	S	R	M	
24. Turbine First Stage Pressure	S	R	M	
25. Logic Channel Testing	N.A.	N.A.	M	
26. Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	M	
27. Control Room Ventilation	N.A.	N.A.	R	Check damper operation for accident mode with isolation signal
28. Control Rod Protection (for use with LOPAR fuel)	N.A.	R	*	

* Within 31 days prior to entering a condition in which the Control Rod Protection System is required to be operable unless the reactor trip breakers are manually opened during RCS cooldown prior to T_{cold} decreasing below 350^{o}F and the breakers are maintained open during RCS cooldown when T_{cold} is less than 350^{o}F.

TABLE 4.1-2

FREQUENCIES FOR SAMPLING TESTS

		Check	Frequency	Maximum Time Between Tests
1.	Reactor Coolant Samples	Gross Activity (1)	5 days/week (1)	3 days
		Radiochemical (2)	Monthly	45 days
		$\bar{E}$ Determination	Semi-annually (3)	30 weeks
		Tritium Activity	Weekly (1)	10 days
		F, Cl & O_2	Weekly	10 days
2.	Reactor Coolant Boron	Boron Concentration	Twice/week	5 days
3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Monthly	45 days
4.	Boric Acid Tank	Boron Concentration	Twice/week	5 days
5.	Boron Injection Tank	Boron Concentration	Monthly	45 days
6.	Spray Additive Tank	NaOH Concentration	Monthly	45 days
7.	Accumulator	Boron Concentration	Monthly	45 days
8.	Spent Fuel Pit	Boron Concentration	Prior to Refueling	NA*
9.	Secondary Coolant	Iodine-131	Weekly (4)	10 days
10.	Containment Iodine-Particulate Monitor or Gas Monitor	Iodine-131 and Particulate Activity or Gross Gaseous Activity	Continuous When Above Cold Shutdown(5)	NA

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TABLE 4.1-2 (Continued)

FREQUENCIES FOR SAMPLING TESTS

FOOTNOTES:

* NA- Not Applicable

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/cc.

(2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 30 minutes making up at least 95% of the total activity of the primary coolant.

(3) $\bar{E}$ determination will be started when the gross analysis indicates $\geq$ 10 μCi/cc and will be redetermined if the primary coolant gross radioactivity changes by more than 10 μCi/cc in accordance with Specification 3.1.D.

(4) When the iodine-131 activity exceeds 10% of the limit in Specification 3.4.A, the sampling frequency shall be increased to a minimum of once each day.

(5) Except as indicated in Specification 3.1.F.

TABLE 4.1-3

FREQUENCIES FOR EQUIPMENT TESTS

	Check	Frequency	Maximum Time Between Tests
1. Control Rods	Rod drop times of all control rods	Each refueling shutdown	**
2. Control Rods	Partial movement of all control rods	Every 2 weeks during reactor critical operations	20 days
3. Pressurizer Safety Valves	Set point	Each refueling shutdown	**
4. Main Steam Safety Valves	Set point	Each refueling shutdown	**
5. Containment Isolation System	Automatic Actuation	Each refueling shutdown	**
6. Refueling System Interlocks	Functioning	Each refueling shutdown prior to refueling operation	NA*
7. DELETED			
8. Diesel Fuel Supply	Fuel Inventory	Weekly	10 days
9. Turbine Steam Stop, Control Valves	Closure	Monthly****	45 days****
10. Cable Tunnel Ventilation Fans	Functioning	Monthly	45 days
11. Control Room and Fuel Handling Building Filtration System	Charcoal Filter Pressure Drop Test < 5 inches of water visual inspection Freon - 112 (or equivalent) test ≥ 99.5% at ambient conditions	Each refueling shutdown prior to refueling operation***	**

Amendment No.

4.16 REACTOR COOLANT SYSTEM AND CONTAINMENT FREE VOLUME LEAKAGE DETECTION AND REMOVAL SYSTEMS SURVEILLANCE

Applicability

Applies to the surveillance and monitoring of leakage detection and removal systems provided for determining and removing reactor coolant leakage and leakage into the containment free volume.

Objective

To verify compliance with operational leakage limits of Specification 3.1.F.

Specifications

A. For the purposes of demonstrating compliance with the operational leakage limits of Specification 3.1.F., the following shall be performed:

1. At least once a shift monitor the leakage detection systems required by Specification 3.1.F.1.a(6).

2. At least once a shift monitor the containment sump inventory and discharge.

3. At least once a shift monitor the recirculation sump inventory and the reactor cavity inventory.

4. At least once daily perform a reactor coolant system water inventory balance.

5. For the RCS/RHR pressure isolation valves, periodic leakage testing(*) shall be accomplished every time the plant is placed in the cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for at least 72 consecutive hours if testing has not been accomplished in the preceding 9 months, and each time any valve may have moved from the fully closed position (any time differential pressure across the valve is less than 150 psig), and prior to returning the valve to service after maintenance, repair or replacement work is performed.

(*) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

B. The containment sump pumps required to be operable by specification 3.1.F.1.a(1) shall be demonstrated to be operable by performance of the following surveillance program:

1. At monthly intervals, each sump pump shall be started and a discharge flow of at least 25 gpm verified.

2. At refueling intervals, each sump pump shall be operated under visual observation to verify that the pumps start and stop at the appropriate setpoints and that the discharge flow is at least 25 gpm per pump.

Basis

These specifications establish the surveillance program for monitoring reactor coolant system leakage and leakage into the containment free volume during plant operation and ensure compliance with Specification 3.1.F. These specifications also establish surveillance requirements for the containment sump pumps. Surveillance requirements for the various leakage detection instrumentation systems are contained in Table 4.1-1 of these specifications.

ATTACHMENT B

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Safety Evaluation

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
March, 1981

Safety Evaluation

The proposed changes contained in Attachment A to this Application would modify the technical specifications to incorporate revised limiting conditions for operation (LCOs) and surveillance requirements regarding leakage detection and removal systems for reactor coolant system leakage and total leakage into the containment free volume. Such technical specification revisions were required by NRC's December 4, 1980 letter from Mr. B. Grier to Mr. E. McGrath (Consolidated Edison) and were committed to in Consolidated Edison's January 5, 1980 response to NRC.

Specifically, technical specifications 1.0 and 3.1.F have been revised to: (1) modify and upgrade the requirements for reactor coolant system leakage detection and removal systems, and (2) establish new requirements for detection of total leakage (both reactor coolant and non-reactor coolant) into the containment free volume. Also, additional requirements have been added to specification 4.1 and a new specification 4.16 has been proposed to establish surveillance requirements for leakage measurement and determination and for the leakage removal systems. To the extent that the NRC's Standard Technical Specifications (STS) for Westinghouse nuclear plants provide guidance for the areas addressed in this Application, the STS were utilized as an initial basis for the present proposed changes. These proposed technical specification revisions in conjunction with the plant modifications that have recently been effected should preclude a recurrence of the October 17, 1980 event which resulted in an excessive accumulation of water inside the Indian Point Unit No. 2 containment building.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee and the Consolidated Edison Nuclear Facilities Safety Committee. Both Committes concur that the proposed changes do not represent a significant hazards consideration and will not cause any change in the types or an increase in the amounts of effluents or any change in the authorized power level of the facility.