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Docket No. 50-247

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. John D. O'Toole
Vice President
Nuclear Engineering and Quality Assurance
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. O'Toole:

The Commission has issued the enclosed Amendment No. 69 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your applications transmitted by letters dated April 25, 1980 and March 26, 1981.

The amendment, in response to the April 25, 1980 application, revises the Technical Specifications to reduce the minimum reactor coolant flow to 95% of thermal design flow and accommodate plant operation with up to 12% of the steam generator tubes plugged. As a separate item, in response to the March 26, 1981 application, the amendment adds to the Technical Specifications limiting conditions for operation and surveillance requirements for leakage detection and removal systems to reduce the likelihood of a recurrence of the October 17, 1980 flooding event.

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

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 69 to DPR-26
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

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concurs as to form subject to changes

Previous concurrences-See next page

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Docket Nos.50-247

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The amendment revises the Technical Specifications to reduce the minimum reactor coolant flow to 95% of thermal design flow, to accommodate plant operation with 12% of the steam generator tubes plugged, and to include limiting conditions for operation and surveillance requirements for leakage detection and removal systems.

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

Sincerely,

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

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RSB:DSI N. Lauben NL 4-9-81

RSB:DSI S.Diab SD 4/21/81

Sent back for rewrite - Needs form of amendment to conform to form of notice JMoore

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Docket

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

April 22, 1981

Docket No. 50-247

Mr. John D. O'Toole
Vice President
Nuclear Engineering and Quality Assurance
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

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Division of Licensing

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

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 69
License No. DPR-26

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

   A. The applications for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated April 25, 1980 and March 26, 1981, comply 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 applications, 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 69 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 22, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 69

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
i	i
ii	ii
1-4	1-4
-------	1-4a
2.1-1	2.1-1
2.1-3	2.1-3
3.1-17 through 3.1-20	3.1-17 through 3.1-23
3.3-15	3.3-15
3.10-1	3.10-1
3.10-9	3.10-9
3.10-11	3.10-11
3.10-16	3.10-16
Figure 3.10-2	Figure 3.10-2a
--------	Figure 3.10-2b
Table 4.1-1 (second sheet)	Table 4.1-1 (second sheet)
Table 4.1-1 (third sheet)	Table 4.1-1 (third sheet)
Table 4.1-2 (both sheets)	Table 4.1-2 (both sheets)
Table 4.1-3 (first sheet)	Table 4.1-3 (first sheet)
--------	4.16-1
--------	4.16-2

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TABLF ꟳ CONTENTS (Continued)

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 <u>UNIDENTIFIED LEAKAGE</u>

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

2 SAFETY LIMTTS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, and coolant temperaure during four-loop and three-loop operation, and reactor coolant flow during four-loop operation.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for four and three-loop operation respectively (additional limitations on three-loop operation are described in section 3.1.A). The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

The Region 1 fuel residence time shall be limited to 21,000 effective full power hours (EFPH) under design operation conditions. The licensee may propose to operate individual assemblies from Region 1 in excess of 21,000 EFPH by providing an analysis which includes the effect of clad flattening or a change in operation conditions. Any such analysis, if proposed, shall be approved by the Regulatory Staff prior to operation in excess of 21.000 EFPH.

The following DNB related parameters pertain to four loop steady state operation at power levels greater than 98% of rated full power (in excess of 2703 MWt):

a. Reactor Coolant System $T_{avg} \leq 573.5^{\circ}F$
b. Pressurizer Pressure $\geq$ 2220 psia
c. Reactor Coolant System Total Flow Rate $\geq$ 340,800 gpm

Item (b), pressurizer pressure, is not applicable during either a thermal power change in excess of 5% of rated thermal power per minute, or a thermal power step change in excess of 10% of rated thermal power.

Under the applicable operating conditions, should reactor coolant temperature, T_{avg}, or pressurizer pressure exceed the values given in items (a) and (b), the parameter shall be restored to its applicable range within 2 hours.

Rod withdrawal block and load runback occurs if reactor trip setpoints are approached within a fixed limit.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in a DNBR of less than 1.30.[4]

The ranges on reactor coolant system temperature, pressure and loop coolant flow [5] during steady-state, four-loop, power operation are specified to assure that the values assumed in the accident analyses are not exceeded during normal plant operation.

Compliance with the specified ranges on reactor coolant system temperature and pressurizer pressure is demonstrated by verifying that the parameters are within their applicable ranges at least once each 12 hours.

Compliance with the specified range on Reactor Coolant System total flow rate is demonstrated by verifying the parameter is within it's range after each refueling cycle.

References

[1]FSAR Section 3.2.2

[2]FSAR Section 3.2.1

[3]FSAR Technical Specification 3.10

[4]FSAR Section 14.1.1

[5]"Analysis and Evaluation of Non-LOCA Transients for Operation with 95% Reactor Coolant System Thermal Design Flow and with 25% Uniform Steam Generator Tube Plugging," dated April, 1980.

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 897A, 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. <u>Total Reactor Coolant System Leakage</u>:

(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. <u>Leakage Into The Containment Free Volume</u>:

(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. 35', 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. 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", 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 view, especially if they are to closed systems, it must be recognized that leaks on 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} μc/cc to 10^{-6} μc/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 1 gpm to 15 gpm per detector can be measured by this system. Condensate flows greater than 15 gpm can be determined using weir calibration curves. Condensate flows 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 predetermined levels, 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 lights 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. 35' or the water level in the reactor cavity reaches EL. 20', the reactor is placed in cold shutdown within the next 36 hours. 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

The seven day out of service period for the Weld Channel and Penetration Pressurization System and the Isolation Valve Seal Water System is allowed because no credit has been taken for operation of these systems in the calculation of off-site accident doses should an accident occur. No other safeguards systems are dependent on operation of these systems.[13] The minimum pressure settings for the IVSWS and WC & PPS during operation assures effective performance of these systems for the maximum containment calculated peak accident pressure of 47 psig.

References

(1) FSAR Section 9

(2) FSAR Section 6.2

(3) FSAR Section 6.2

(4) FSAR Section 6.3

(5) FSAR Section 14.3.5

(6) FSAR Section 1.2

(7) FSAR Section 8.2

(8) FSAR Section 9.6.1

(9) FSAR Section 14.3

(10) Indian Point Unit No. 2, "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10CFR50.46 and Appendix K of 10CFR50", dated December 1978, and "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10CFR50.46 and 10 CFR Part 50, Appendix K," dated April, 1980.

(11) Letter from William J. Cahill, Jr. of Consolidated Edison Company of New York, to Robert W. Reid of the Nuclear Regulatory Commission, dated July 13, 1976. Indian Point Unit No. 2 Small Break LOCA Analysis.

(12) Indian Point Unit No. 3 FSAR Sections 6.2 and 6.3 and the Safety Evaluation accompanying "Application for Amendment to Operating License" sworn to by Mr. William J. Cahill, Jr. on March 28, 1977.

(13) FSAR Sections 6.5 and 6.6.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the limits on core fission power distributions and to the limits on control rod operations.

Ojbectives:

To ensure:

1. Core subcriticality after reactor trip,

2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and

3. Limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications:

3.10.1 Shutdown Reactivity

The shutdown margin shall be at least as great as shown in Figure 3.10-1.

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

(a) $F^N_{\Delta H} \leq 1.55 \left[1 + 0.2\ (1-P)\right]$

(b) For $\leq$ 6% steam generator tube plugging:

$F_Q(Z) \leq (2.31/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.62) \times K(Z)$ for $P \leq .5$

(c) For $>$ 6% but $\leq$ 12% steam generator tube plugging:

$F_Q(Z) \leq (2.25/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.50) \times K(Z)$ for $P \leq .5$

where P is the fraction of full power at which the core is operating; K(Z) is the fraction given in Figure 3.10-2a (for $\leq$ 6% tube plugging) or Figure 3.10-2b (for $>$ 6% but $\leq$ 12% tube plugging); and Z is the core height location of F_Q.

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

The upper bound envelope of the total peaking factor (F_Q) of specification 3.10.2.1 times the normalized peaking factor axial dependence of Figures 3.10-2a and b has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on the specified F_Q times the normalized envelope of Figures 3.10-2a and b indicate a peak clad temperature of less than 2200°F for the double-ended cold leg guillotine break with C_D=0.6, the worst case break.(1) (2)

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_{\Delta H}$ there is a 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g. rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system.

to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that the total peaking factor upper bound envelope of specified F_Q times Figures 3.10-2a and b is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted
with the control rod bank more than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of ±5 percent ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference. Figure 3.10-5 shows a typical construction of the target flux difference band at BOL and Figure 3.10-6 shows the typical variation of the full power value with burnup.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 4 week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

The required drop time to dashpot entry is consistent with safety analysis.

REFERENCE

1. Indian Point Unit No. 2, "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10 CFR 50.46 and Appendix K of 10 CFR 50." See also Consolidated Edison Company's letter to NRC dated January 5, 1979 which submitted the results of this reanalysis based on the Westinghouse ECCS Evaluation Model approved by NRC letter to Westinghouse dated August 29, 1978.

2. Indian Point Unit No. 2, "Analysis of the Emergency Core Cooling System in accordance with the acceptance criteria of 10CFR50.46 and 10 CFR Part 50, Appendix K," dated April, 1980.

Figure 3.10-2 a

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE
(For S.G. tube plugging levels up to 6%)

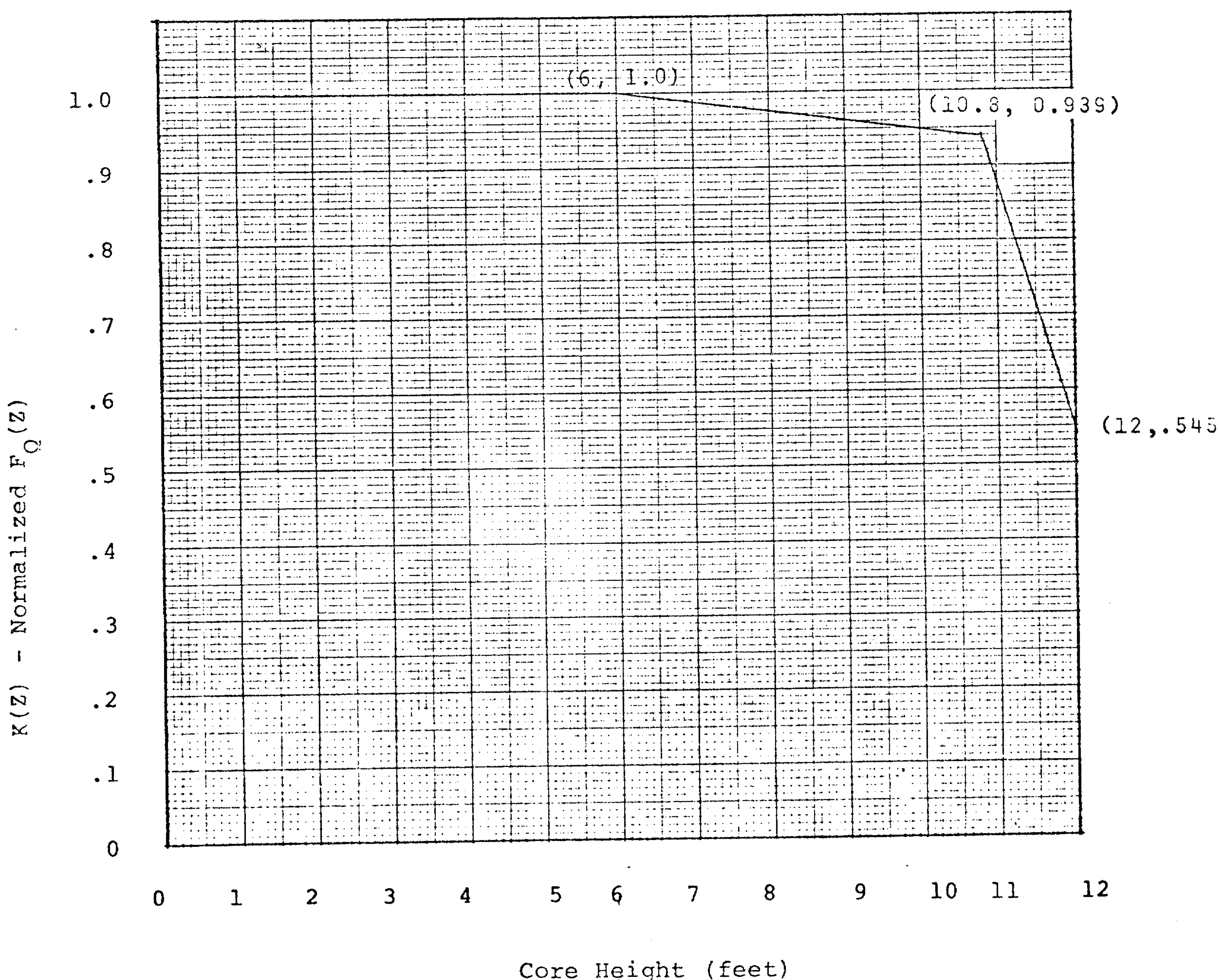
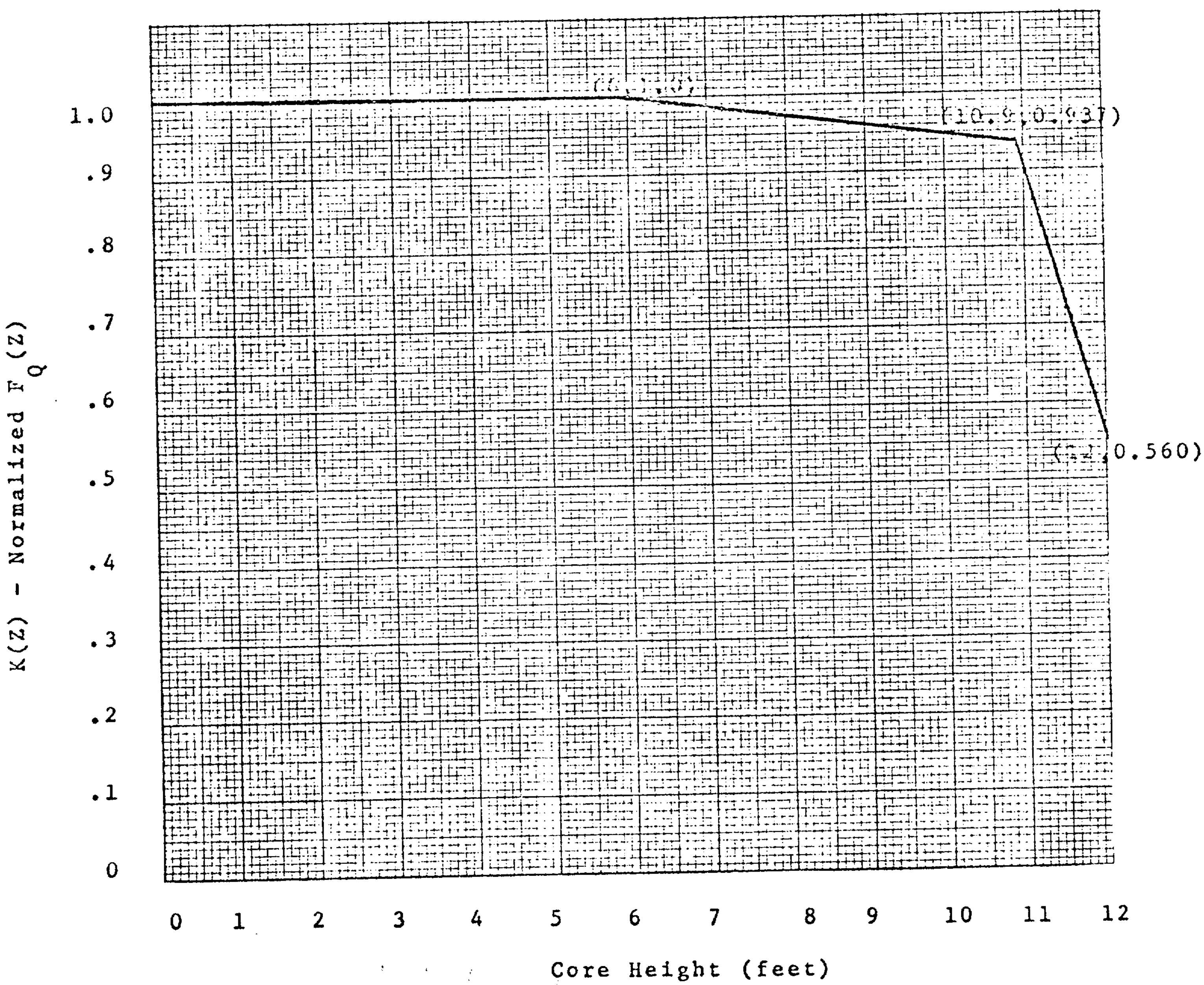


Figure 3.10-2b

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE
(For S.G. tube plugging levels > 6% but ≤ 12%)



Amendment No. 69

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 calibratio
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°F and the breakers are maintained open during RCS cooldown when T_{cold} is less than 350°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

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***	**

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 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. Minimum test differential pressure shall not be less than 150 psid.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 69 FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

Introduction

By letter dated April 25, 1980, the Consolidated Edison Company of New York (the licensee) proposed an amendment to the Technical Specifications for Indian Point, Unit No. 2. The proposal reduces the minimum reactor coolant flow to 95% of the thermal design flow and revises the limit for total nuclear peaking factor (F_Q) to accommodate plant operation with up to 12% of the steam generator tubes plugged. (At present, less than 4% of the steam generator tubes are plugged).

By letter dated March 26, 1981, the licensee proposed other Technical Specification changes to include limiting conditions for operation and surveillance requirements for leakage detection and removal systems. These were developed to 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.

Evaluation

95% Flow, 12% Plugging

With the April 25, 1980 application, the licensee submitted the results of a loss-of-coolant accident (LOCA) reanalysis for the limiting break size, assuming up to 12% of the steam generator tubes are plugged. By letter dated February 20, 1981, the licensee provided additional information to justify that the limiting break size was unchanged for the 12% plugging case.

The reanalysis demonstrates that with a maximum total nuclear peaking factor (F_Q) of 2.25 and up to 12% plugging, ECCS performance will meet the acceptance criteria of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. We conclude that the proposed change in F_Q in the Technical Specifications is acceptable.

The licensee also evaluated the impact on non-LOCA transients of operation with 95% reactor coolant system thermal design flow and up to 25% of the steam generator tubes plugged. The most severe loss of flow transient was

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reanalyzed with these assumptions and the minimum DNBR remains above 1.30. We have reviewed the licensee's evaluation of the non-LOCA transients and conclude that the 5% reduction in thermal deisgn flow and plugging of up to 25% of the steam generator tubes does not result in a violation of the safety limits. Thus, the proposed change in minimum reactor coolant system flow in the Technical Specifications is acceptable.

Leakage Detection and Removal Systems

By letter of March 26, 1981, the licensee proposed Technical Specification changes to include limiting conditions of operation (LCO) and surveillance requirements for leakage detection and removal systems. We compared these to the Westinghouse Standard Technical Specifications (STS) and found them to be at least as conservative. Furthermore, several provisions of the licensee's proposal, such as LCO's and surveillance requirements for the sump pumps and limits on total leakage into containment, are not included in the STS.

In addition, LCO's and surveillance requirements were proposed for the Reactor Coolant System/Residual Heat Removal pressure isolation valves. Although testing of these valves was a requirement in the February 11, 1980 Confirmatory Order (Item A.5 of Appendix), the proposed Technical Specification clarify the requirements.

We conclude that the changes proposed in the March 26, 1981 application are acceptable, and should reduce the likelihood of recurrence of the October 17, 1980 flooding event.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 22, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-247

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 69 to Facility Operating License No. DPR-26, issued to the Consolidated Edison Company of New York, Inc. (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 2 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of the date of issuance.

The amendment, in response to the April 25, 1980 application, revises the Technical Specifications to reduce the minimum reactor coolant flow to 95% of thermal design flow and accommodate plant operation with up to 12% of the steam generator tubes plugged. As a separate item, in response to the March 26, 1981 application, the amendment adds to the Technical Specifications limiting conditions for operation and surveillance requirements for leakage detection and removal systems to reduce the likelihood of a recurrence of the October 17, 1980 flooding event.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environemtal impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated April 25, 1980 and March 26, 1981, (2) Amendment No. 69 to License No. DPR-26, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the White Plains Public Library, 100 Martine Avenue, White Plains, New York. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 22 day of April, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing