March 7, 1990

Docket No. 50-247

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SNewberry

Mr. Stephen B. Bram
Vice President, Nuclear Power
Consolidated Edison Company
of New York
Broadway and Bleakley Avenue
Buchanan, New York 10511

Dear Mr. Bram:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 69542)

The Commission has issued the enclosed Amendment No. 148 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to License Condition 2.C.(1) Maximum Power Level and to the Technical Specifications in response to your application transmitted by letter dated September 30, 1988, as supplemented on January 10, 1989, March 30, 1989, April 14, 1989, October 19, 1989, January 4, 1990, and February 8, 1990.

The amendment revises the Indian Point Nuclear Generating Unit No. 2 Operating License and Technical Specifications to increase its rated thermal power from 2758 Megawatts-thermal (MWt) to 3071.4 MWt. This amendment will increase the electrical output of Indian Point 2 by approximately 115 Megawatts-electrical.

The Advisory Committee on Reactor Safeguards (ACRS) reviewed your application during its 358th meeting on February 9, 1990. A copy of the ACRS's February 15, 1990 letter concerning your application is enclosed.

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

Sincerely,
ORIGINAL SIGNED BY:
Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 148 to DPR-26
2. Safety Evaluation
3. Letter from Carlyle Michelson, Chairman, ACRS, to Kenneth C. Carr, Chairman, NRC, dated February 15, 1990

cc: w/enclosures
See next page

* SEE PREVIOUS CONCURRENCE

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| LA:PDI-1 | PM:PDI-1 | SRXB:NRR* | PRPB* | EMEB* | SPLB* | EMTB* |
| CVogan | DBrinkman:rsc | RJones | LCunningham | LMarsh | CMcCracken | CYCheng |
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Mr. Stephen B. Bram
Vice President, Nuclear Power
Consolidated Edison Company
of New York
Broadway and Bleakley Avenue
Buchanan, New York 10511

Dear Mr. Bram:

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The amendment revises the Indian Point Nuclear Generating Station Unit No. 2 Operating License and Technical Specifications to increase its rated thermal power from 2758 Megawatts-thermal (MWt) to 3071.4 MWt. This amendment will increase the electrical output of Indian Point 2 by approximately 115 Megawatts-electrical.

The Advisory Committee on Reactor Safeguards (ACRS) reviewed your application during its 358th meeting on February 9, 1990. A copy of the ACRS's February 15, 1990 letter concerning your application is enclosed.

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

Sincerely,

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. to DPR-26
2. Safety Evaluation
3. Letter from Carlyle Michelson, Chairman, ACRS, to Kenneth C. Carr, Chairman, NRC, dated February 15, 1990

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Mr. Stephen B. Bram
Consolidated Edison Company
of New York, Inc.

Indian Point Nuclear Generating
Station 1/2

cc:

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Buchanan, New York 10511

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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. 148
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated September 30, 1988, as supplemented on January 10, 1989, March 30, 1989, April 14, 1989, October 19, 1989, January 4, 1990, and February 8, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by a change to paragraph 2.C.(1) Maximum Power Level and by changes to the Technical Specifications as indicated in the attachment to this license amendment; paragraphs 2.C.(1) and 2.C.(2) of Facility Operating License No. DPR-26 are hereby amended to read as follows:

   (1) Maximum Power Level

   The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3071.4 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 148, 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 and is to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 7, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 148

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
| 1-1 | 1-1 |
| 2.3-2 | 2.3-2 |
| 3.1.G-1 | 3.1.G-1 |
| 3.4-1 | 3.4-1 |
| 3.4-2 | 3.4-2 |
| 3.8-2 | 3.8-2 |
| 3.8-5 | 3.8-5 |

# TECHNICAL SPECIFICATIONS

## 1 DEFINITIONS

The following used terms are defined for uniform interpretation of the specifications.

1.1 a. Rated Power

A steady state reactor thermal power of 3071.4 MWT.

b. Thermal Power

The total core heat transfer rate from the fuel to the coolant.

1.2 Reactor Operating Conditions

1.2.1 Cold Shutdown Condition

When the reactor is subcritical by at least 1% $\Delta k/k$ and $T_{avg}$ is $\leq 200°F$.

1.2.2 Hot Shutdown Condition

When the reactor is subcritical, by an amount greater than or equal to the margin as specified in Technical Specification 3.10 and $T_{avg}$ is $> 200°F$ and $\leq 555°F$.

1.2.3 Reactor Critical

When the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

1.2.4 Power Operation Condition

When the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

(3) Low pressurizer pressure - $\geq$1870 psig.

(4) Overtemperature $\Delta T$

$$\Delta T \leq \Delta T_o \ [(K_1 - K_2 (T - T') + K_3 (P - P') - f( I)]$$

where:
$\Delta T$ = Measured $\Delta T$ by hot and cold leg RTDs, °F
$\Delta T_o$ $\leq$ Indicated $\Delta T$ at rated power

$T$ = Average temperature, °F
$T'$ = Design full power Tave at rated power, $\leq$579.7°F
$P$ = Pressurizer pressure, psig
$P'$ = 2235 psig
$K_1$ $\leq$ 1.25
$K_2$ = 0.022
$K_3$ = 0.00095

and f($\Delta$I) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -36% and +7%, f ($\Delta$I) = 0, where $q_t$ and $q_b$ are percent RATED POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total POWER in percent of RATED POWER;

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -36%, the $\Delta T$ Trip Setpoint shall be automatically reduced by 2.14% of its value at RATED POWER; and

(iii) For each percent that the magnitude of $q_t - q_b$ exceeds +7%, the $\Delta T$ Trip Setpoint shall be automatically reduced by 2.15% of its value at RATED POWER.

(5) Overpower $\Delta T$

$$\Delta T \leq \Delta T_o \ [K_4 - K_5 \frac{dT}{dt} - K_6 \ (T - T'')]$$

where:
$\Delta T$ = Measured $\Delta T$ by hot and cold leg RTDs, °F

$\Delta T_o$ $\leq$ Indicated $\Delta T$ at rated power

$T$ = Average temperature, °F

$T''$ = Indicated full power $T_{avg}$ at rated power $\leq$ 579.7°F

$K_4$ $\leq$ 1.074

$K_5$ = Zero for decreasing average temperature

$K_5$ $\geq$ 0.188, for increasing average temperature (sec/°F)

$K_6$ $\geq$ 0.0015 for $T \geq T''$; $K_6 = 0$ for $T < T''$

$\frac{dT}{dt}$ = Rate of change of $T_{avg}$

## G. REACTOR COOLANT SYSTEM PRESSURE, TEMPERATURE, AND FLOW RATE

### Specifications

The following DNB related parameters pertain to four loop steady-state operation at power levels greater than 98% of rated full power:

a. Reactor Coolant System $T_{ave} \leq 587.2°F$

b. Pressurizer Pressure $\gtrsim$ 2190 psia

c. Reactor Coolant System Total Flow Rate $\gtrsim$ 331,840 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.

### Basis

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 the safety limit DNBRs.

The limits on reactor coolant system temperature, pressure and loop coolant flow represent those used in the accident analyses and are specified to assure that the values assumed in the accident analyses are not exceeded during steady-state four loop operation. Indicator uncertainties have not been accounted for in determining the DNB parameter limits on temperature and pressure.

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.

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the Steam and Power Conversion System.

Objective

To define conditions of the turbine cycle steam-relieving capacity. Auxiliary Feedwater System and City Water System operation is necessary to ensure the capability to remove decay heat from the core.

Specification

A. The reactor shall not be heated above 350°F unless the following conditions are met:

(1) A minimum ASME code approved steam relieving capability of twenty (20) main steam valves shall be operable (except for testing). With up to three (per steam generator) of the twenty main steam line code-approved safety relief valves inoperable, heat-up above 350°F and power operation is permissible provided either the inoperable valve(s) is restored to operable status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.4-1.

(2) Three auxiliary feedwater pumps, each capable of pumping a minimim of 380 gpm, must be operable.

(3) A minimum of 360,000 gallons of water in the condensate storage tank and a backup supply from the city water supply.

(4) Required system piping, valves, and instrumentation directly associated with the above components operable.

(5) The main steam stop valves are operable and capable of closing in five seconds of less.

(6) The total iodine activity of I-131 and I-133 on the secondary side of the steam generator shall be less than or equal to 0.15 uCi/cc.

## Basis

Reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condensers. Thereafter, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to feed the steam generators is provided by operation of the turbine cycle feedwater system.

The operability of the twenty main steam line code safety valves ensure that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% Rated Thermal Power coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The total relieving capacity of the twenty main steam safety valves is 15,108,000 lbs/hr which is 114 percent of the total secondary steam flow of 13,310,000 lbs/hr at 100% NSSS Power (3083.4 Mwt). Startup and/or power operation is allowable with main steam safety valves inoperable within the limitations of Table 3.4-1 on the basis of the reduction in secondary system steam flow and Thermal Power required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following basis:

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP= Reduced reactor trip setpoint in percent of RATED THERMAL POWER

V= Maximum number of inoperable safety valves per steam line

(109)= Power Range Neutron Flux-High Trip Setpoint for (4) loop operation

X= Total relieving capacity of all safety valves per steam line (3,777,000 lbs/hr).

Y= Maximum relieving capacity of any one safety valve (823,000 lbs/hr).

In the unlikely event of complete loss of electrical power to the station, decay heat removal would continue to be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps, and steam discharge to the atmosphere via the main steam safety valves and

6. The requirements for RHR pump and heat exchanger operability/operation in Specifications 3.8.A.3 and 3.8.A.4 may be suspended during maintenance, modification, testing, inspection, repair or the performance of core component movement in the vicinity of the reactor pressure vessel hot legs. During operation under the provisions of this specification, an alternate means of decay heat removal shall be available when the required number of RHR pump(s) and heat exchanger(s) are not operable. With no RHR pump(s) and heat exchanger(s) operating, the RCS temperature and the source range detectors shall be monitored hourly.

7. The reactor Tavg shall be less than or equal to 140°F.

8. Specification 3.6.A.1 shall be adhered to for reactor subcriticality and containment integrity.

B. With fuel in the reactor vessel and when:

i) the reactor vessel head is being moved, or
ii) the upper internals are being moved, or
iii) loading and unloading fuel from the reactor, or
iv) heavy loads greater than 2300 lbs (except for installed crane systems) are being moved over the reactor with the reactor vessel head removed,

the following specifications (1) through (12) shall be satisfied:

1. Specification 3.8.A above shall be met.

2. The minimum boron concentration shall be $\geq$2000ppm and shall provide a shutdown margin $\geq$ 5% $\Delta k/k$. The required boron concentration shall be verified by chemical analysis daily.

3. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

4. No movement of fuel in the reactor shall be made until the reactor has been subcritical for at least 174 hours.

The shutdown margin requirements will keep the core subcritical.

During refueling, the reactor refueling cavity is filled with borated water. The minimum boron concentration of this water is the more restrictive of either 2000ppm or else sufficient to maintain the reactor subcritical by at least 5% $\Delta k/k$ in the cold shutdown condition with all rods inserted. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses.

Periodic checks of refueling water boron concentration ensure the proper shutdown margin. The specifications allow the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are utilized during refueling to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The 174 hour decay time following plant shutdown and the 23 feet of water above the top of the reactor vessel flanges are consistent with the assumptions used in the dose calculations for fuel-handling accidents both inside and outside of the containment. The analysis of the fuel handling accident inside of the containment is based on an atmospheric dispersion factor ($X/Q$) of $5.1 \times 10^{-4}$ sec/m$^3$ and takes no credit for removal of radioactive iodine by charcoal filters. The requirement for the fuel storage building charcoal filtration system to be operating when spent fuel movement is being made provides added assurance that the offsite doses will be within acceptable limits in the event of a fuel-handling accident. The additional month of spent fuel decay time will provide the same assurance that the offsite doses are within acceptable limits and therefore the charcoal filtration system would not be required to be operating.

The requirement that at least one RHR pump and heat exchanger be in operation ensures that sufficient cooling capacity is available to maintain reactor coolant temperature below 140°F, and sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR pumps and heat exchangers operable when there is less than 23 feet of water above the vessel flange ensures that a single failure will not result in a complete loss of residual heat removal capability. With the head removed and at least 23 feet of water above the flange, a large heat sink is available for core cooling, thus allowing

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

## 1.0 INTRODUCTION

By letter dated September 30, 1988, as supplemented January 10, 1989, March 30, 1989, April 14, 1989, October 19, 1989, January 4, 1990, and February 8, 1990, the Consolidated Edison Company of New York, Inc. (the licensee), requested an amendment to Facility Operating License No. DPR-26 for the Indian Point Unit 2 plant. Specifically, the amendment would revise the Indian Point 2 Operating License and Technical Specifications to increase the rated core power level from the present core licensed power level of 2758 Megawatts thermal (MWt) to a core licensed power level of 3071.4 MWt. The proposed change would allow Indian Point 2 to operate at a Nuclear Steam Supply System (NSSS) power level of 3083.4 MWt. The proposed change represents an approximate 11.3 percent increase over the presently licensed core power rating of 2758 MWt. The proposed change would increase the unit's gross electrical output from approximately 906 Megawatts electrical (MWe) to approximately 1021 MWe.

The licensee's submittals of October 19, 1989, January 4, 1990, and February 8, 1990 provided supplemental information to the original submittal dated September 30, 1988. Thus these submittals did not alter the action as noticed in the Federal Register on May 31, 1989, or affect the proposed no significant hazards consideration.

This increase does not represent any change in equipment or equipment design rating. The original plant design and construction were based upon a guaranteed core thermal power level of 3071.4 MWt and an NSSS power level of 3083.4 MWt. The original license request, however, was only for operation at a core thermal power level of 2758 MWt even though our evaluation of the engineered safety features (with the exception of the emergency core cooling system) and our accident analyses were performed for a core thermal power level of 3216 MWt. In a letter dated October 19, 1989, the licensee stated that there were no technical bases for limiting the requested core power level to 2758 MWt, but, that since Indian Point 2 was the first Westinghouse four-loop plant of its design to seek an operating license, the licensee elected the current power level, which was based on scale-ups from previously licensed plants, to permit the accumulation of experience before operation at the vendor guaranteed power level of 3071.4 MWt.

During this review, the staff reviewed information submitted in the original application dated September 30, 1988 and supplemental letters dated January 10, 1989, March 30, 1989 (which includes WCAP-11972, Revision 1 [Consolidated Edison Company of New York, Inc., Indian Point Unit 2, NSSS Stretch Rating - 3083.4 MWt, Licensing Report]), April 14, 1989, October 19, 1989, January 4, 1990, and February 8, 1990. The licensee's submittal of January 20, 1989 on the incorporation of Westinghouse Optimized Fuel Assemblies, WCAP-12003, Revision 1 (Westinghouse Improved Thermal Design Procedure Instrument Uncertainty Methodology for Consolidated Edison Company of New York - Indian Point 2) and WCAP-8567-P-A (Improved Thermal Design Procedure), and the Updated Final Safety Analysis Report (FSAR) were also consulted as part of the review.

## 2.0 EVALUATION

### 2.1 Nuclear Steam Supply System (NSSS)

The scope of the licensee's review to support the proposed core uprating encompassed all aspects of the Indian Point 2 NSSS design and operation affected by the increase. In addition, a review was conducted by the licensee to identify any potential unreviewed safety questions that might occur as a result of the increased power level in accordance with 10 CFR 50.59. The review encompassed the verification that NSSS components and systems will continue to meet functional requirements specified in the Updated FSAR for the increased power level. Current NRC-approved analytical techniques were used for certain reanalyses performed at the increased power level. These are noted in the applicable sections of this evaluation.

## Core Design

By letters dated September 30, 1988, December 30, 1988, and January 20, 1989, the licensee described a fuel design transition to Optimized Fuel Assembly (OFA) loading for Indian Point Unit No. 2. These submittals considered operation at the increased core power level of 3071.4 MWt. During the Cycle 10 (spring 1989) reload approximately 35 percent of the core was loaded with OFAs. The remaining portion of the core is low-parasitic (LOPAR) fuel. Both fuel types were reviewed for a core power of 3071.4 MWt with respect to nuclear design, thermal-hydraulic design, and fuel performance. The effects of mix core operation were also evaluated. Current NRC-approved analytical techniques were used for the analyses which were performed at the increased power rating.

The staff reviewed and approved the use of the LOPAR and OFA fuel at 2758 MWt when it approved the reload and Technical Specification changes for Cycle 10 (Amendment No. 140 to Facility Operating License No. DPR-26, dated May 18, 1989).

This approval was based on a staff review of the licensee's analyses performed at 3071.4 MWt, which bounded the Cycle 10 proposal. The Cycle 10 safety assessment was done in anticipation of the power upgrade proposal and is therefore acceptable.

## Auxiliary Feedwater and Residual Heat Removal

In its September 30, 1988 proposal, the licensee stated that transient analyses at the increased power level had been performed for the following events involving the auxiliary feedwater system (AFS): (1) loss of normal feedwater and (2) loss of all ac power to station auxiliaries. The first of these events is the limiting transient with regard to minimum AFS flow requirements for the Indian Point 2 AFS design. An increase in the AFS flow was assumed in these analyses in order to assure that sufficient heat removal capability exists to prevent pressurizer filling and water relief from the pressurizer relief or safety valves. The licensee therefore proposed a Technical Specification change identifying the increased AFS flow rate. The proposed Technical Specification change is consistent with the flow assumed in the supporting transient analyses and is, therefore, acceptable.

The staff has also compared the Indian Point 2 AFS flow capacity against cooling requirements for the uprated power and concluded that cooldown time to residual heat removal (RHR) system cut-in conditions would not be significantly affected.

Based on the above the staff agrees with the licensee that the impact of the power uprating on AFS and RHR performance would not be significant.

## Emergency Core Cooling System (ECCS)

The licensee's study concluded that no adverse impact on ECCS operability, or additional vulnerabilities to single failure would result from the power uprating. ECCS performance for the uprated power was demonstrated in analyses supporting the Cycle 10 reload (approved by Amendment No. 140 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2, May 18, 1989). These analyses were reviewed by the staff and found to demonstrate compliance with 10 CFR 50.46 and Appendix K. The analysis for a large break LOCA was performed assuming a core thermal power of 102 percent of 3071.4 MWt, an F-DELTA-H of 1.62, 25 percent uniform steam generator tube plugging and a reduced core flow (90 percent of original design). The large and small break LOCA analyses were performed with current NRC-approved analytical methods. The assumptions match the conditions for the proposed power upgrade; therefore, the Indian Point 2 ECCS is adequate for the proposed power upgrade and associated Technical Specification changes.

The original licensing-basis time for hot leg switchover to long-term RHR cooling following a LOCA was based on an analysis performed for a core power of 3425 MWt. The results of the switchover calculation presently presented in the Indian Point 2 FSAR are, therefore, bounding for the proposed power upgrade.

Accident Analysis

The licensee indicated that both the LOCA analyses (small break and large break) and all of the FSAR Chapter 14 non-LOCA analyses were reviewed to determine their continued acceptability based upon plant operations at the stretch power conditions. The following non-LOCA analyses were reanalyzed or evaluated at the stretch power conditions.

a. Uncontrolled Rod Cluster Control Assembly Withdrawal from a Subcritical or Low Power Startup Condition

b. Uncontrolled Rod Cluster Control Assembly Withdrawal at Power

c. Rod Cluster Control Assembly Drop

d. Chemical and Volume Control System Malfunction

e. Loss of Reactor Coolant Flow

f. Locked Rotor

g Rupture of a Control Rod Mechanism Housing - Rod Cluster Control Assembly Ejection

h. Loss of Normal Feedwater

i. Loss of All AC Power to the Station Auxiliaries

j. Steam Line Break Mass and Energy Release Inside Containment

k. Startup of an Inactive Reactor Coolant Loop

l. Loss of External Electrical Load

m. Reduction in Feedwater Enthalpy

n. Excessive Load Increase

o. Steam Line Break - Core Response

The results of the LOCA analyses (also see Section 2.2) and results of the analyses for events a., c., d., and g. above are contained in the Cycle 10 analyses (approved by Amendment No. 140 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2, May 18, 1989). The staff reviewed the Cycle 10 analyses and concluded that the appropriate safety criteria were met. Since the Cycle 10 analyses were performed for a core power of 3071.4 MWt, an increased F-DELTA-H of 1.62, a reduced core flow (90 percent), and steam generator tube plugging of up to 25%, the staff finds that these analyses continue to be applicable and acceptable for operation at the uprated power level.

Events b., e., and f. were reanalyzed after the Cycle 10 OFA review was completed. The reanalysis showed that the DNBR limit value was not violated for these transients at the uprated power. The staff, therefore, finds the results acceptable.

Events h., i., m., n., and o. were also analyzed and found acceptable by the licensee. The confirmatory analyses used assumptions consistent with the proposed Technical Specification changes, and were performed using currently-approved analysis methodology and resulted in parameter values within the safety limits. The staff therefore finds the results acceptable.

Event l. was analyzed assuming a 100 percent load rejection resulting from a turbine trip with concurrent loss of main feedwater. No credit is taken for operation of reactor coolant system relief valves, steam line relief valves, steam dump system, pressurizer level control system, pressurizer spray, or direct reactor trip on turbine trip. The results of this analysis demonstrate that the combined capacity of the three pressurizer safety valves is adequate to meet the identified performance criterion (prevent the pressurizer pressure from exceeding 110 percent design pressure) when credit is taken for the first safety-grade signal (high pressurizer pressure) from the reactor protection system. Since this is consistent with the original design basis for the plant, we find the design remains acceptable for operation at core powers up to 3071.MWt (NSSS power of 3083.4 MWt).

The analysis results of event j. are contained in Section 2.2 and are acceptable.

Event k. was considered in the original design bases for the plant. However, subsequent to initial plant operation, a change to the allowable plant operating conditions was made to prohibit power operation with a loop out of service (i.e., N-1 loop operation). The current Technical Specifications require that all 4 reactor coolant pumps be operating for reactor power operation and prohibits operation with an inactive loop. Therefore, since N-1 loop operation is prohibited at power, the startup of an inactive reactor coolant loop event as considered in the original plant design bases is precluded and reanalysis of this event was not performed or required.

By letter dated March 30, 1989, the licensee provided supplemental information related to reactor coolant system pressure and temperature Technical Specification changes. The March 30, 1989 submittal included a revision to Technical Specification page 3.1.G-1 which is the revision the staff reviewed and for which this approval applies.

Reactor Coolant System

The licensee performed an assessment for the impact of the stretch power level on the Reactor Coolant System (RCS) loop analysis. It was indicated that the increased temperature range, as a result of stretch rated power, would not have significant impact on the loop piping and supports as originally designed. In addition, stresses and fatigue usage for the components of the reactor vessel upper and lower internals were evaluated for the revised design transients for the stretch rating conditions and found to be within acceptable limits.

Based on the NRC approval of Leak-Before-Break (LBB) technology for Indian Point Unit 2 (letter dated February 23, 1989, to S. Bram, Con Ed, from D. Brinkman, NRC, with attached safety evaluation), the large RCS loop breaks were eliminated from consideration. For the LOCA condition, the break sizes for consideration after LBB are significantly less severe than the original design basis breaks. The new total loads on the RCS will, therefore, also be equal to or less than the existing loads.

NSSS Summary

Based on its review of the power upgrade transient and accident reanalysis and referencing of past review of OFA fuel core reloads (License Amendment No. 140) within the scope of the systems areas discussed above, the staff finds the proposed Indian Point 2 power uprating to 3071.4 MWt thermal power (3083.4 MWt NSSS power) acceptable for steam generator tube plugging up to a maximum of 25 percent.

## 2.2 Containment Integrity Analysis

The licensee performed a long term loss-of-coolant accident (LOCA) analysis for the uprated NSSS conditions to determine the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large break LOCA. The licensee analyzed the LOCA mass and energy release using the Westinghouse analytical model (Topical Report WCAP-10325, "Westinghouse LOCA Mass and Energy Release Model for Containment Design"). A full double-ended reactor coolant pump suction break with minimum engineered safety features was determined to be the most limiting case for a hypothetical rupture of the reactor coolant pipe. The computer codes described in Topical Report Report WCAP-10325 were used to generate the mass and energy releases for this limiting case for both the 3083.4 MWt and the 3216 MWt power levels and the proposed operating temperature ranges.

The COCO computer code (Topical Report WCAP-8327, 1974) was used to generate containment pressure and temperature responses. Consistent with the mass and energy release analysis, the failure of one diesel generator to start and the availability of three containment fan coolers and one containment spray pump was assumed to be the minimum engineered safety feature condition. The containment model assumed no credit for recirculation spray. The peak containment pressure was calculated to be 40.31 psig and 41.12 psig for the 3083.4 MWt and 3216 MWt power levels, respectively, compared to the containment design value of 47 psig.

The licensee also analyzed the consequences of a postulated main steam line break (MSLB) in order to determine the acceptability of the safety-related containment systems to mitigate the consequences of a hypothetical rupture of a main steam pipe. The LOFTRAN computer code (Topical Report WCAP-7907-P-A, 1984) was used to generate the mass and energy released to the containment for a large double ended MSLB at full power. The following assumptions were made for the MSLB analysis: (1) failure of one feedwater control valve concurrent with the break which results in additional mass and energy release to the containment, (2) full reactor coolant flow with availability of offsite power throughout the transient without coastdown of the reactor coolant pumps, (3) minimum safety injection with failure of a diesel generator, and (4) isolation of the main feedwater flow to the faulted loop by closure of the feedwater pump discharge valves at 60 seconds following receipt of a safety injection signal.

The COCO computer code was used to generate the containment pressure and temperature responses for the MSLB analysis. The containment model was identical to that for the long term LOCA analysis. Minimum engineered safety features (three fan coolers and one spray pump) and failure of one feedwater control valve were assumed in the MSLB analysis. The peak containment pressure as a result of the MSLB was calculated to be 39.99 psig. This value is less than that for the LOCA and thus containment integrity is governed by the long-term LOCA analysis.

The staff has reviewed the licensee's accident analyses for containment performance and found the assumptions and calculated results to be conservative. The analytical methodology of the LOFTRAN and COCO computer programs, and the computer codes described in Topical Report WCAP-10325, used for these analyses have been previously reviewed and found acceptable by the staff. Therefore, the staff finds the containment integrity analyses for the power uprating conditions to be acceptable.

## 2.3 Balance of Plant Systems

The licensee evaluated major BOP component and system designs to verify their compliance with the functional requirements specified in the Updated FSAR when the plant is operated at the stretch rated power of 3083.4 MWt. The licensee evaluated the following BOP systems/components:

(1) Steam turbine system

The steam turbine system was designed for a maximum flow rate of 13,940,000 lb/hr at an inlet steam pressure of 730 psia corresponding to a thermal power rating of 3216 MWt.

(2) Main feedwater system

The main feedwater system was designed to supply sufficient feedwater flow to the four steam generators when they are operating at a thermal load of 3083.4 MWt.

(3) Main steam and reheat steam system

The capability of the main steam system and components to operate at 3083.4 MWt is enveloped by the original design. The safety relief valve capacities are sufficient to support operation at the stretch rated power level. The staff's evaluation of the steam generators is provided in Section 2.6.

(4) Main moisture separator/reheater

The moisture separator/reheater system provides low pressure steam to the low pressure turbines and to the main feed pump turbines. The major components of the system were verified to be adequate to operate at the stretch rated power level of 3083.4 MWt.

(5) Main turbine/generator auxiliary systems

The auxiliary systems supporting the turbine/generator which includes the lube oil, main turbine control oil, generator hydrogen cooling, generator seal oil, and generator stator cooling, were verified to be capable of supporting operation at the stretch rated power level.

(6) Steam gland sealing system

The gland sealing system was designed to prevent the leakage of steam from the turbine cylinders at the maximum power level of 3216 MWt.

(7) Extraction steam, heater drain and vents

These extraction steam components take steam from the turbine cycle at various stages. The design parameters for operation at the stretch rated power level are bounded by the original plant design.

(8) Condenser air removal system

The condenser air removal system was designed to support condenser operation at the stretch rated power of 3083.4 MWt.

(9) Circulating water system

The circulating water system capacity was verified to be suitable for operation at the stretch rated power of 3083.4 MWt.

(10) Condensate system

The condensate system was verified to be adequate for operation at the stretch rated power of 3083.4 MWt.

(11) Condensate and feedwater chemical feed system

The condensate and feedwater chemical feed system was verified to be capable of supporting operation of the condensate and feedwater systems at the stretch rated power level.

(12) Boiler feed pump lube oil system

The two boiler feed pumps and their auxiliaries are each sized for 60% of full load operation (i.e., 3083.4 MWt), and therefore, can be operated at the stretch rated power level.

(13) Steam generator blowdown system

The existing steam generator blowdown system was evaluated for a power level of 3083.4 MWt and found to be adequate to maintain the secondary side water chemistry within required specifications.

(14) Primary makeup water system

The existing primary makeup water system provides adequate capacity at the stretch rated power level for the reactor coolant system and other systems that require demineralized water.

(15) Essential service water system

The new containment integrity analyses performed at a power level of 3216 MWt assumed reduced cooling requirements for the containment fan cooler units by approximately 20%. This has resulted in added margin for the essential service water system under design basis LOCA conditions. Therefore, the system has adequate margin for supporting safe plant shutdown under accident conditions. The system was also determined to provide sufficient flow to support power operation at the uprated level.

(16) Non-essential service water system

The various balance-of-plant heat loads served by the non-essential service water system do not require flow immediately following design basis accidents. The system was determined to provide sufficient flow to support power operation at the uprated level.

(17) Station and instrument air systems

Plant operation at the stretch rated power level has no effect on performance of the station and instrument air systems.

(18) Secondary sampling system

No additional secondary sampling will be required for operation at the stretch rated power level. Slight changes in various fluid parameters (temperature and pressure) at the uprated conditions are within the sampling system capability.

(19) Containment design

The containment integrity analysis performed at a power level of 3216 MWt determined a peak containment pressure of 41.12 psig for the bounding accident (LOCA) which is within the containment design pressure limit of 47 psig. Refer to Section 2.2 for additional discussion.

(20) Main control room panels

The main control room panels are not affected by the stretch rated power level.

(21) Miscellaneous control panels for equipment and systems

The miscellaneous control panels are not affected by the stretch rated power level.

(22) Heating, ventilating and air conditioning (HVAC) system

The HVAC systems for the control room, fuel storage building, and containment (purge and pressure relief) are not affected by the stretch rated power level.

(23) Emergency diesel generator

The safety related power loads supplied by the emergency diesel generators were examined and determined not to be affected by the stretch rated power level.

(24) Balance of Plant (BOP) Piping Systems

For BOP piping systems, the licensee also performed a detailed review of original piping stress analysis. This review confirmed that the pipe stresses for the range of temperatures and pressures at stretch rated conditions would remain within the applicable code allowable limits.

A review of the existing analysis on BOP High Energy Line Break (HELB) was also performed by the licensee. It showed that the HELB analysis would not be affected by operation under stretch rating conditions.

The staff has reviewed the above BOP systems and found that they were either originally designed for a power level of 3083.4 MWt or 3216 MWt, or found to be adequate for supporting plant operation and post-accident safe shutdown from the stretch rated power level. Therefore, the staff finds the licensee's evaluation of the BOP systems and components including the piping attached to steam generators to be acceptable.

### 2.4 Equipment Qualification

The licensee evaluated the effects of the proposed stretch rated power on equipment qualification (EQ) and confirmed that the equipment in the EQ program per 10 CFR 50.49 is qualified for the temperature, pressure and radiation levels corresponding to harsh environments from postulated pipe break accidents at the stretch power of 3083.4 MWt. The licensee confirmed that the temperature profile previously approved for equipment qualification bounds the environmental conditions resulting from a LOCA or MSLB at the higher power level. Consequently, the staff finds the licensee's equipment qualification evaluation to be acceptable.

### 2.5 Instrumentation

Since this submittal contained no changes to plant equipment and hence no changes to plant control, safety or instrumentation equipment (hardware), our review of the plant's instrumentation concentrated on locating and evaluating any changes in instrumentation setpoints made necessary by changes in operating or safety limits and to assure that such changes were accommodated by the instrumentation within the constraints of the plant setpoint methodology described in WCAP-8567 and WCAP-12003, Revision 1. The only significant instrumentation and control setpoint changes were to the Overtemperature-Delta T (OT-Delta T) and Overpower-Delta T (OP-Delta T) setpoint channels. Potential concerns with uncertainties associated with reactor power determination, pressurizer pressure measurement, RCS flow measurement, the OT-Delta T and OP-Delta T setpoints were investigated. Our review found that the setpoints had been determined and the uncertainties analyzed according to the statistical methods developed as part of the Westinghouse Improved Thermal Design Procedure discussed in WCAP-12003, Revision 1. These methods have been approved by the staff; therefore, we find them acceptable.

As part of its submittal, the licensee reviewed the suitability of their BOP instrumentation and control equipment for service at the proposed power level. We have reviewed the licensee's findings and our conclusions are as follows.

Main Control Room Panels - The licensee states that no additional equipment which requires control from the Main Control Room is required for operation at the proposed power level and that no changes to the existing Main Control Room Panels are needed. Our review confirms this finding and we conclude that the current layout remains acceptable.

Miscellaneous Control Panels - The licensee states that operation at the proposed power level does not require the addition or modification of equipment controlled from these panels and no changes to these panels are needed. Our review confirms this finding and we conclude that the current layout remains acceptable.

Plant Annunciator - The licensee states that operation at the proposed power level does not require the addition or modification of equipment which requires annunciation and no changes to the annunciator system are needed. Our review confirms this finding and we conclude that the current layout of the annunciator system remains acceptable.

120 Volt A-C Instrument Power - The licensee states that these four safety-related busses were originally designed to supply the instrumentation required for operation at 3071.4 MWt core power. They further state that operation at this proposed power level does not require the addition of new of increased loads to this system and no changes are needed. Our review confirms this finding and we conclude the current configuration of these busses remains acceptable.

125 Volt D-C Power System - The licensee states that operation at the proposed power level does not require the addition of new or increased loads to this system and no changes are needed. Our review confirms this finding and we conclude that the current configuration remains acceptable.

Load Shedding and Emergency Load Sequencing - The licensee states that operation at the proposed power level does not require the addition of new or increased loads to the emergency busses and that no modifications or changes to the current load shedding and subsequent reloading schemes, control equipment and instrumentation are needed. Our review confirms this finding and we conclude that the current schemes, control equipment and instrumentation remain acceptable.

### Instrumentation Summary

On the basis of its review of the information contained in the licensee's submittals and other documents identified in Section 1.0 of this report, the staff has concluded that the design and configuration of the instrumentation and control equipment at Indian Point 2 remains acceptable for operation at the proposed power level. The staff also concludes the revised setpoints of the OP-Delta T and OT-Delta T trips were determined in accordance with approved methodologies and the setpoint uncertainties for the reactor protection system were likewise determined using approved methodology and are acceptable for operation at the proposed power level.

## 2.6 Steam Generators

### 1987 Refueling Outage

During a scheduled inservice inspection (ISI) of the steam generator (SG) #22 girth weld (upper transition cone to upper shell weld), ultrasonic reflectors were detected on the inside weld circumference. Cracks were confirmed by an entry into the secondary side manway for visual examination, magnetic particle (MT) inspection and localized grinding. The inside surface contained general corrosion pitting and closely spaced intermittent linear indications over a large extent of the circumference. The linear indications observed were predominantly in the vicinity of the heat affected zones of the weld. Ultrasonic testing (UT) and MT inspections were then performed on the other steam generators. These inspections revealed similar, although less severe, surface indications.

The licensee developed detailed repair procedures. After the initial MT, grinding was permitted to a 1/4" depth. In the event that cracks were still present, grinding continued at 1/8" increments followed by a visual examination or MT after each increment. After the indications were cleared, the excavation was contoured and mapped. A post repair ultrasonic and magnetic particle test was performed to confirm that no surface indication remained. Corrosion pits outside the repair area were not removed.

The licensee provided a stress and fatigue analysis to support the final weld configuration after repair. The evaluation addresses local minimum shell thickness and stress and fatigue considerations. The evaluations use detailed finite element analyses and simple mechanics formulations for the loading condition defined in the steam generator Design Specification. The analyses considered grindout configurations 3/4 inch and one inch deep with additional local grindouts. The purpose was to conservatively envelope the actual grinding pattern. The area of deepest penetration has a land section of four inches and two inches, respectively. The blend-out to the shell inner surface contains a 2:1 taper at each end of the land. A 0.5 inch radius is assumed at the blend of the taper to the land. Local grinds could be of any length circumferentially up to and including the total circumference and multiple grinds were acceptable.

Eight boat samples were removed for metallurgical analysis by three laboratories. The results were as follows:

(1) The cracks initiated at surface pits in or near the weld.

(2) The cracks propagated transgranularly, generally in the heat affected zone (HAZ).

(3) The pits and cracks were corrosion filled with the corrosion product primarily $Fe_3O_4$, but containing Zn, Cu, and a trace of S.

(4) Hardness was greater than 30 Rc in HAZ. This indicates little if any tempering from the post weld heat treatment.

The licensee concluded that the propagation was typical of corrosion fatigue. This condition occurs in fossil boiler drums and was similar to observations at Indian Point Unit 3 and the Surry Power Station.

In a safety evaluation dated January 8, 1988, the staff reached the following conclusions:

(1) The licensee used a technically acceptable approach to repair the girth weld by processive grinding of the cracks to established final weld profiles.

(2) The licensee removed all observed linear indications.

(3) The conditions that caused crack initiation and propagation may still be present.

(4) The licensee should perform certain augmented inservice inspections to confirm that cracks have not re-initiated.

1989 Refueling Outage

A magnetic particle examination was conducted initially on 1/3 of the inside circumference of the SG #22 girth weld, pursuant to the licensee's letter to the NRC dated December 11, 1987. Linear indications were detected during this examination. Subsequently 100% of the inside circumferences of the girth welds in all four steam generators were inspected. Linear indications were also detected in these additional examinations. All observed cracks were ground out and weld repairs with post weld heat treatment were accomplished on SG #22.

The probable mechanism of the occurrence of the cracks has been established as corrosion fatigue caused by a combined action of thermal cycling, oxygen in the auxiliary feedwater and copper from the feedwater system. To minimize the cause for the thermal fatigue mechanism, the licensee removed the downcomer flow resistance plates. Additionally, the controls of the feedwater low flow valves have been modified and the licensee is evaluating operating changes.

A visual examination conducted on the SG #22 feedwater nozzle inside radius section in accordance with the Inservice Inspection Program detected several linear indications within the nozzle inner radius section in the bottom 120° segment of the nozzle. Visual and liquid penetrant tests were subsequently performed on the nozzles of all four steam generators. SG #22 and #23 nozzles contained similar cracks whereas none were detected in SG #21 and #24. During the course of performing liquid penetrant tests on the nozzles, linear indications were also detected in support bracket welds directly below the nozzle inside radius section on SG #22 and #23. No such indications were detected in SG #21 and #24. The indications on the SG #22 and #23 nozzle inner radius were removed by grinding and the indications on the SG #22 and #23 support bracket welds were removed and repair welded.

A detailed description of the licensee's inspection results and repair program is available in the non-proprietary report, WCAP-12294, Revision 1, entitled "Indian Point Unit 2 Steam Generator Girth Weld/Feedwater Nozzles Report, Spring, 1989 Outage." The staff's review of this document concluded that the final girth weld configuration was encompassed by the original safety evaluation dated January 8, 1988.

Operation at the Proposed Increased Power Level

As described in the Introduction, the original plant design and construction was based upon the proposed increased power level. Although the licensee ordered and has taken delivery of four Model 44F replacement steam generators, the licensee has not established a schedule for the installation of these replacement units.

The licensee has reviewed the balance of plant systems and components to determine the impact of increasing the power level and concluded that the installed steam generators are adequate for the proposed new operating conditions. In a letter dated May 26, 1989, the licensee committed to shut down for a mid-cycle outage at which time an inspection of steam generators #22 and #23 girth welds will be performed. The inspection results will be shared with NRC staff. This mid-cycle outage began on February 24, 1990.

The causal mechanism for the cracking of the steam generator girth weld has not been conclusively established. The removal of the downcomer flow resistance plate during the 1989 refueling outage should be an improvement to minimize the thermal cycling. The environment and conditions that caused crack initiation and propagation in the girth weld may still be present. Therefore, the staff will evaluate the licensee's inspection results from the mid-cycle outage.

<u>Steam Generators Summary</u>

A. The staff intends to closely monitor the licensee's operation and inservice inspection of the installed steam generators until the repaired upper transition cone welds are removed from service.

B. The staff determined that the proposed increase in power level will not result in any significant increase in the susceptibility of cracking in the upper transition cone weld.

C. The staff will evaluate the licensee's inspection results and the staff will assure that flaw initiation and growth will be within acceptable limits during each period of operation.

## 2.7 Power Level-Related Requirements Levied Since Issuance of Original Operating License

Many new requirements have been levied on licensees since the original Indian Point 2 operating license was issued to Consolidated Edison in 1973. Some of the new requirements were power level-related. Imposition of the new requirements was from various sources which included new or revised NRC regulations, NRC Bulletins, and Generic Letters.

In its February 8, 1990 submittal, the licensee stated that it has had in effect, since it received the Indian Point 2 operating license, a general practice to recognize and accommodate for purposes of plant evaluations, the likelihood of future operation at up to the NSSS maximum calculated power level of 3216 MWt. The licensee's FSAR updating process and its reload safety analysis process are designed to provide current up-to-date reference to plant design and analyses. Documents generated by these activities were used as key inputs to determine areas where additional power level-related analyses and evaluations were required to be performed. The licensee also conducted separate reviews of its safety evaluations performed under 10 CFR 50.59, Generic Letters and NRC Bulletins for power level impacts. The licensee determined that the potential power level impacts of these reviews had been either properly included in the stretch power level submittals, or will require only setpoint changes, procedural changes, or revisions to internal documentation. The licensee also stated that any currently active issues covered by Generic Letters or NRC Bulletins will be addressed at the stretch power level.

In addition to the licensee's activities and reviews described in its February 8, 1990 submittal, the staff independently audited four power level-dependent issues. The four issues were the station blackout (SBO) rule (10 CFR 50.63), the anticipated transients without scram (ATWS) rule (10 CFR 50.62), NRC Bulletin 88-02 "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes", and Generic Letter 81-21 "Natural Circulation Cooldown". The licensee's April 14, 1989 response to the SBO rule considered operation at the stretch power level of 3071.4 MWt. Although the staff has not yet completed its review of the licensee's SBO response, this review will consider operation at 3071.4 MWt; therefore, the SBO issue will be properly dealt with. The ATWS issue has also been properly dealt with in that the staff has reviewed and approved the licensee's ATWS mitigation system (letter from Donald S. Brinkman, NRC, to Stephen B. Bram, Con Ed, dated May 16, 1989). The staff's review of the

Indian Point 2 ATWS mitigation system was included in a generic review of Westinghouse-designed ATWS mitigation systems. The review is applicable to operation of Indian Point 2 at 3071.4 MWt. The ATWS mitigation system was installed and made operational during the refueling outage which ended on July 2, 1989. The staff also confirmed that the licensee's analysis and actions taken in response to Bulletin 88-02 were applicable at 3071.4 MWt as was the generic analysis for natural circulation cooldown. Therefore, based on the staff's review of the licensee's review program described in its February 8, 1990 letter, and the staff's review of the above noted four issues, the staff concludes that adequate consideration has been given to power level-dependent issues not specifically addressed in the licensee's proposal to increase the licensed thermal power at Indian Point 2.

## 2.8 Environmental Consequences

### Radiological

The licensee's original operating license safety analysis of radiological source terms for normal operation and accidents was performed for an assumed power level of 3216 MWt. This power level bounds the proposed power level of 3071.4 MWt. The staff review for the present proposal relative to radiological consequences was limited to one Design Basis Accident reassessed by the licensee, the Steam Generator Tube Rupture (SGTR) accident. The licensee reassessment consisted of a sensitivity study on the impact of the parameter change associated with the higher power rating. Based on its review of the reassessment, the staff concludes that (1) the offsite dose increases assessed by the licensee due to the higher power rating are insignificant, (2) the total offsite doses previously calculated by the licensee were well within the acceptance criteria delineated in Standard Review Plan Section 15.6.3, and (3) the licensee's reassessment does not alter our conclusions stated in Section 11.4 of the Indian Point Unit No. 2 SER. We find that the licensee's dose calculations and their assumptions are reasonable and are, therefore, acceptable.

Non-Radiological

The original licensing evaluations for the plant, including the Final Environmental Impact Statement related to operation of Indian Point Nuclear Generating Plant Unit No. 2, September 1972 were based on a design thermal output of 3216 MWt corresponding to the NSSS safeguards analysis. Therefore, the proposed uprating remains within the bounds of the original environmental analyses.

2.9 Technical Specification Changes

The licensee identified the following Technical Specification changes related to the power uprating:

(1) Page 1-1, Section 1.1.a - Change the value of rated power from 2758 to 3071.4 MWt. This change identifies the increase in core power rating and is consistent with the power level assumed in the supporting analyses.

(2) Page 2.3-2, Section 2.3.1.B and page 3.1.G-1, Section 3.1.G - Change the allowable vessel average temperature to 587.2°F (page 3.1.G-1 in the March 30, 1989 supplemental submittal). This increase is acceptable for the uprated power level provided that the T-average in the overtemperature and overpower delta-T setpoint equations is increased to maintain comparable plant protection to bound the reactor core safety limits. The proposed value of T-average in the setpoint equations (579.9°F) is identified on page 2.3-2 of the proposed TS changes. The value was determined using present NRC-approved methodology, provides the necessary margin, and is therefore acceptable.

(3) Page 3.4-1, Section 3.4.A.2 - The minimum required auxiliary feedwater pump capacity is increased to 380 gpm per pump to reflect increased RHR requirements for the higher power level. This value is consistent with the flow assumed in supporting transient analyses.

(4) Page 3.4-2, Section 3.4 - This Bases section is changed to identify the NSSS power level of 3083.4 MWt which corresponds to the core thermal power plus net reactor coolant pump heat.

(5) Page 3.8-2, Section 3.8.B.4 - The minimum decay time before fuel can be moved in the reactor has been increased to 174 hours. This value is consistent with the decay time assumed in the fuel handling accident analysis.

Certain other Technical Specification values based on analyses at the uprated power level (e.g., F-DELTA-H and system pressure) have been previously identified, reviewed by the staff, and approved for the Indian Point Unit 2 Cycle 10 OFA reload.

The Technical Specification revisions correspond with the assumptions used in the supporting confirmatory analyses or previously approved bounding analyses, therefore, the staff finds them acceptable.

## 3.0 ENVIRONMENTAL CONSIDERATIONS

Pursuant to 10 CFR 51.21, 51.32, and 51.35 an Environmental Assessment and Finding of No Significant Impact has been prepared and was published in the Federal Register on December 6, 1989 (54 FR 50459). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

## 4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

Dated: March 7, 1990

PRINCIPAL CONTRIBUTIONS:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 15, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED POWER LEVEL INCREASE FOR INDIAN POINT NUCLEAR GENERATING STATION UNIT 2

During the 358th meeting of the Advisory Committee on Reactor Safeguards, February 8-10, 1990, we reviewed the application of Consolidated Edison Company of New York (Licensee) for a license amendment, to permit it to operate the Indian Point Nuclear Generating Station Unit 2 at a core thermal power level up to 3071.4 MWt. The current core power level limit is 2758 MWt, so this is approximately an 11 percent increase. This matter was discussed by our Subcommittee on the Systematic Assessment of Experience, on February 6, 1990. During these meetings, we had the benefit of discussions with representatives of both the NRC staff and the Licensee. We also had the benefit of the documents referenced. The NRC staff recommends approval of this application.

The plant was originally licensed in 1973, at a core thermal power level up to 2758 MWt, though the original analyses and supporting environmental assessments, with the exception of the emergency core cooling system (ECCS), were made for a core thermal power level of 3216 MWt. The ECCS was evaluated at 2758 MWt. There is nothing in the history to suggest that the lower power level of the original license was based on anything other than a (commendable) caution, since this was the first of the large Westinghouse 4-loop plants to seek a license. Since this is a license amendment, the staff review is based on the original license requirements, and our review is confined to the implications of the proposed power level increase, not to a review of the original license decision.

Since nearly all the original analyses were performed at the higher power, the remaining need was to demonstrate ECCS operability at the proposed power, and this was done in May of 1989. The analyses were reviewed by the NRC staff, and found to be in compliance with

10 CFR 50.46 and Appendix K, with suitable conservatism. We have no reason to question these conclusions.

Many new requirements, not all due to the Three Mile Island accident, have been levied since the original license was issued in 1973. Some of these are power related, and the staff should assure itself that those will be met at the new power level. The Licensee has assured the NRC, in a letter dated February 8, 1990, that that is the case.

Subject to resolution of this matter to the satisfaction of the NRC staff, we believe that the Indian Point Nuclear Generating Station Unit 2 can be operated at core power levels up to 3071.4 MWt without undue risk to the health and safety of the public.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:

1. Memorandum dated January 29, 1990 from S. A. Varga, Nuclear Regulatory Commission, to R. F. Fraley, ACRS, Subject: Transmittal of Revision to Draft Safety Evaluation to Increase Licensed Thermal Power Level of the Indian Point Nuclear Generating Unit No. 2
2. Letter dated September 30, 1988 from S. Bram, Consolidated Edison Company to U. S. Nuclear Regulatory Commission transmitting Application for Amendment to Operating License (Indian Point Station Unit 2)
3. Letter dated February 8, 1990 from S. Bram, Consolidated Edison Company to Donald S. Brinkman, Nuclear Regulatory Commission, Subject: Application for License Amendment to Increase Authorized Power Level