C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(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. 135, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

D.(1) Steam Generator Inspections

The plant shall be brought to the cold shutdown condition within sixteen equivalent months of operation from August 31, 1979, but in any event, no later than May 1, 1981. For the purpose of this requirement, equivalent operation is defined as operation with a reactor coolant temperature greater than 350°F. An inspection of all four steam generators shall be performed and Nuclear Regulatory Commission approval shall be obtained before resuming power operation following this inspection.

D.(2) Secondary Water Chemistry Monitoring

The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

(a) Identification of a sampling schedule for the critical parameters and control points for these parameters;

(b) Identification of the procedures used to quantify parameters that are critical to control points;

(c) Identification of process sampling points;

(d) Procedure for the recording and management of data;

(e) Procedures defining corrective actions for off control point chemistry conditions; and

(f) A procedure identifying the authority responsible for the interpretation of the data, and the sequence and timing of administrative events required to initiate corrective action.

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% Δ 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 ΔT

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

where:
ΔT = Measured ΔT by hot and cold leg RTDs, °F
ΔT_o $\leq$ Indicated ΔT 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(ΔI) 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 (ΔI) = 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 ΔT 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 ΔT Trip Setpoint shall be automatically reduced by 2.15% of its value at RATED POWER.

(5) Overpower ΔT

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

where:
ΔT = Measured ΔT by hot and cold leg RTDs, °F

ΔT_o $\leq$ Indicated ΔT 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 $T_{ave} \leq 585.5°F$

b. Pressurizer Pressure $\geq$ 2206 psig

c. Reactor Coolant System Total Flow Rate $\geq$ 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

Attachment B

Safety Assessment

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
September 30, 1988

SAFETY ASSESSMENT

The Indian Point Unit No. 2 stretch rating analysis was performed to verify that for an increase in the Nuclear Steam Supply System (NSSS) licensed operating power from 2770 MWt to 3083.4 MWt, the most limiting FSAR accident events, equipment, systems, and fuel design features would remain in compliance with the current licensing basis of the Indian Point 2 plant. These accident analyses, systems and equipment were reviewed in detail with respect to the stretch rating impact on the following aspects of plant design and operation:

1. Consequences of accidents postulated in the Final Safety Analysis Report (FSAR).

2. Capability of systems and equipment to meet design bases specified in the FSAR.

3. Capability of equipment to maintain structural integrity under conditions defined in the FSAR.

4. Definition of Nuclear Steam Supply System/Balance of Plant (NSSS/BOP) safety-related interfaces.

5. Operating limits and conditions contained in Technical Specifications impacted by the power rating increase and operation over a range of temperatures, pressures and tube plugging levels.

This review demonstrates that based on performance under the most limiting analyses and evaluations Indian Point Unit 2 is capable in its present design configuration of operating safely at the stretch rated NSSS power of 3083.4 MWt and that at such a power level the most limiting aspects of the systems and components of the plant remain in compliance with the current licensing requirements for Indian Point Unit 2. The non-limiting aspects of the plant have been evaluated and based on the plant design and analysis on record they will also remain in compliance with the current licensing requirements for Indian Point Unit 2. Confirmatory analysis of the non-limiting evaluations will be submitted as needed to support this amendment request. NSSS and BOP systems and components have been reviewed and found to be able to operate at the uprated power within their design capability. The safety design bases and power generation design bases as defined in the FSAR will also be met under the uprated power conditions. The review also indicates that with Unit operation at 3083.4 MWt:

1. The probability of an accident previously evaluated in the FSAR will not be increased.

2. The consequences of an accident previously evaluated in the FSAR will not be increased.

3. The possibility of an accident which is different from those already evaluated in the FSAR will not be created.

4. The probability of a malfunction of NSSS equipment important to safety, previously evaluated in the FSAR, will not be increased.

5. The consequences of a malfunction of NSSS equipment important to safety, previously evaluated in the FSAR, will not be increased.

6. The possibility of a malfunction of NSSS equipment important to safety, different from any already evaluated in the FSAR, will not be created.

7. The margin of safety as defined in the bases of any Technical Specification will not be reduced.

Based upon the foregoing, it has been concluded that operation of Indian Point Unit 2 at the stretch rated power level of 3083.4 MWt over the full load vessel average temperature range from 549°F to 579.7°F does not involve a significant hazards consideration as defined in 10 CFR 50.92(c).

Large Break LOCA, Small Break LOCA, and various limiting non-LOCA transients have been reanalyzed in conjunction with the separate licensing submittal to allow the use of the optimized fuel assemblies at Indian Point Unit 2 at the stretch rated NSSS power level of 3083.4 MWt (see Con Edison submittal dated September 30, 1988). Additional limiting analyses and evaluations of non-limiting transients were performed to support the stretch rating and are presented in Enclosures 1 and 2 to Attachment B. The non-limiting transients were evaluated in sufficient detail to allow a best engineering judgment to be reached that they are conservatively bounded by the limiting analyses. Confirmatory analyses of the non-limiting transients will be submitted as needed to support this amendment request. Evaluations were also made to confirm that existing analyses relating to containment integrity and radiological accidents, which were performed at 3216 MWt, bound the proposed uprated power level.

NSSS SYSTEMS AND COMPONENTS EVALUATION

The limiting aspects of the NSSS systems and components were evaluated to determine the impacts of increased power level on Indian Point Unit 2. The review indicates that all safety, functional, and structural criteria as defined in the FSAR will be met at the stretch rated power level. The review was performed in accordance with the following guidelines (put forth in WCAP-10263, entitled "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant":

1. The evaluation encompasses all aspects of the Unit's system design and operation, and NSSS equipment mechanical design, which are impacted by the power increase.

2. The evaluation was performed in accordance with the licensing criteria and standards which currently apply to the Unit.

3. Equipment mechanical designs were evaluated against the original industry design codes and standards to which the equipment was built.

4. Current analytical techniques have been used for analyses required in the course of the NSSS review.

Evaluation of the limiting aspects of the NSSS equipment designs that are impacted by the increased power level shows that in most cases, requirements for operation at the higher power are enveloped by either the original Indian Point Unit 2 design or by the generic component design, or have been found to be acceptable by analysis. In a few cases it was necessary to perform additional design transients to verify component capability for operation in compliance with the original design codes and standards at the stretch rated conditions. In every case, it has been established that the limiting aspects of existing NSSS equipment are such that Indian Point Unit 2 is capable of safe and reliable operation at 3083.4 MWt. Based on the operational test, however, it may be necessary to modify the present steam generators for improved moisture removal to meet HP turbine moisture inlet criteria.

Review of the limiting aspects of the NSSS systems design and operating capability indicates that the NSSS will remain in compliance with the functional requirements specified in the FSAR if Indian Point 2 is operated at the increased power rating. A few system parameters and setpoints will need to be revised to reflect operation at the higher power, but those operating conditions have been shown to be within the capability of the relevant NSSS systems as they currently exist. Although an increased power level would be achievable through the use of preexisting operating margins in the NSSS, BOP and Engineered Safeguards Systems, more than ample design and operating margins would be retained to assure safe and reliable operation of the plant.

In connection with this application we have reviewed the most recent results from steam generator inspections conducted during the 1987 refueling outage (see Bram letter dated January 22, 1988, and NRC's safety evaluation dated September 23, 1988). The percentage of plugged steam generator tubes at Indian Point Unit 2 is currently 7%, which is among the lowest number of plugged tubes at currently operating units. Based upon current steam generator conditions and a review of the thermal, hydraulic and pressure changes associated with operation at 3083.4 MWt, we have concluded that stretch power operation will not have an adverse impact upon expected steam generator service life or increase the likelihood of steam generator tube rupture.

Under increased power level, Technical Specification changes will be required for the Auxiliary Feedwater System. The effect of higher decay heat results in higher feedwater flow requirements, which are within the original auxiliary feedwater capability. Although not requiring a technical specification change, the RHR System would also be impacted; as a result of higher decay heat associated with the stretch rating, plant cooldown times will increase slightly. However, even with such an increase, safety requirements would still be satisfied with respect to single train cooldown under accident conditions. The systems that were reviewed are listed in Table 1.

BOP SYSTEMS AND COMPONENTS EVALUATION

As a part of the Indian Point Unit No. 2 stretch rating study, an evaluation of the balance of Plant (BOP) systems and components was conducted to demonstrate that the unit is capable of safe, reliable operation at increased rating of 3083.4 MWt. The evaluation confirmed that Indian Point Unit 2 is capable of operating at the proposed license rating with adequate margin. The evaluation is presented in Enclosure 2 to Attachment B of this Application. The systems that were reviewed are listed in Tables 2 and 3 of Enclosure 2.

NUCLEAR FUEL EVALUATION

The effect on the fuel design of a power level uprating to 3071.4 MWt core power was evaluated. Both the fuel currently in the core (LOPAR) and reload OFA fuel were reviewed at 3071.4 MWt with respect to nuclear design, thermal hydraulic design, and fuel performance. The effects of mixed core operation were also evaluated. Results of this review indicate there are no significant effects on the fuel design due to the stretch rating. Refer to Con Edison submittal dated September 30, 1988 for a detailed description of the fuel design review.

ACCIDENT AND RADIOLOGICAL EVALUATIONS

The LOCA and the limiting non-LOCA analyses presently in the FSAR were reanalyzed to demonstrate that all applicable safety criteria would be met for the stretch rated conditions. These analyses were performed at an NSSS power of 3083.4 MWt. Results show that all applicable safety criteria are met.

The LOCA analyses submitted by letter dated September 30, 1988, which support LOCA and OFA fuel were performed at an NSSS power of 3083.4 MWt, and the parameters assumed in those analyses are consistent with the stretch rating. Results show that all applicable safety criteria are met.

The non-LOCA analyses presently in the FSAR were either reanalyzed or evaluated for stretch rated conditions. These reanalyzed limiting non-LOCA cases were performed at an NSSS power of 3083.4 MWt and the assumed parameters are consistent with the stretch rating. Results show that all applicable safety criteria are met. Engineering evaluations indicate that the non-limiting cases would be conservatively bounded by the cases which were reanalyzed. Confirmatory analyses of the non-limiting cases will be submitted as previously indicated.

The licensing basis supported by this report is for 25% maximum steam generator tube plugging. The radiological source terms were found to be bounded by presently docketed analysis in the FSAR performed at 3216 MWt.

RELEASES DURING NORMAL PLANT OPERATION

Indian Point Unit 2 plant releases and offsite doses were previously evaluated by the Staff for a rated reactor power level of 3216 MWt in the original SER dated November 16, 1970, and thus bound the current application.

ENVIRONMENTAL EVALUATION

Con Edison has performed an environmental evaluation to support its license amendment application to adjust the Indian Point Unit 2 authorized power level from 2758 MWt to a core thermal power of 3071.4 MWt, corresponding to a NSSS power level of 3083.4 MWt. Compared to the current authorized power level, the requested power level would not provide a significant adverse environmental impact, a significant increase in effluents, nor an adverse impact on land, water or cultural resources. With respect to these impacts, the original licensing evaluations for Indian Point Unit 2, including the Atomic Energy Commission's Safety Evaluation dated November 16, 1970, and its September 1972 and February 1975 Final Environmental Statements for the Indian Point site, together with Con Edison's Final Facility Description and Safety Analysis Report as amended pursuant to 10 CFR § 50.71(e), were either based upon a combined Unit 1 and Unit 2 NSSS power level in excess of 3216 MWt, or were based upon, or currently assume, a NSSS thermal power level for Unit 2 alone of 3216 MWt. Accordingly, for purposes of assessing environmental impacts, the requested power level is bounded by the Unit's original environmental licensing analyses and the updated Final Safety Analysis Report (FSAR).

BASIS FOR NO SIGNIFICANT HAZARDS EVALUATION

This evaluation supports Consolidated Edison Company of New York's license amendment request for an increase in the authorized thermal power level of Indian Point Unit 2. At present, Indian Point Unit 2 is licensed to operate at a reactor core thermal power level of 2758 MWt, which corresponds to a Nuclear Steam Supply System (NSSS) power level of approximately 2770 MWt. This amendment requests the necessary license and Technical Specification changes to operate the Unit at a core thermal power level of 3071.4 MWt, which corresponds to an NSSS thermal power level of 3083.4 MWt. This represents an increase of 11.3%. This rating, when combined with the latest ECCS Analysis for 3083.4 MWt, is bounded by the original safety evaluation dated November 16, 1970 for an Engineered Safety Features Design Rating of 3216 MWt. A detailed Basis for No Significant Hazards Evaluation for each proposed revision to the Technical Specifications is presented in Enclosure 3 to Attachment B.

TABLE 1

INDIAN POINT UNIT 2 STRETCH SYSTEM REVIEW

NSSS RELATED SYSTEMS AND COMPONENTS

DESCRIPTION
Reactor Vessel & Internals
Primary Piping
Reactor Coolant System
Auxiliary Feedwater
Safety Injection
Auxiliary Coolant (CCW, RHR, SFP)
Containment Spray
Waste Disposal (Liquid/Gas)
IVSWS
Nuclear Sampling
Chemical & Volume Control
Reactor Protection
Loose Parts Monitoring
Incore Neutron Monitoring
Ex-core Neutron Monitoring
Reactor Control
Containment Spray
Containment Integrity
Containment Atmosphere (Purge/Relief)
Containment Hydrogen Control
Control Rod Drive
Solid Radwaste

ENCLOSURE 1

to

ATTACHMENT B

Safety Assessment

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
September 30, 1988

ENCLOSURE 2

to

ATTACHMENT B

Balance of Plant Systems Evaluation

Consolidated Edison Compa ıy of New York, Inc.
Indian Point ⁚ ıit No. 2
Docket No. 50-247
September 3 , 1988

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT UNIT 2

BALANCE OF PLANT

STRETCH RATING - 3083.4 MWt

LICENSING REPORT

September 1988

UNITED ENGINEERS & CONSTRUCTORS INC.
30 South 17th Street
Post Office Box 8223
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CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT 2

BOP STRETCH RATING - 3083.4 MWt

LICENSING REPORT

TABLE OF CONTENTS

I. INTRODUCTION

Indian Point Unit 2 (IP-2) is currently licensed to operate at a core power level of 2758 MWt. Consolidated Edison conducted a "stretch rating" effort to increase the electrical output of IP-2 to a level corresponding to a Nuclear Steam Supply System (NSSS) output of 3083.4 MWt.

As a part of the Indian Point Unit 2 stretch rating study, UE&C conducted a review of the major Balance of Plant (BOP) component and system designs to verify that they will remain in compliance with the functional requirements specified in the FSAR when Indian Point Unit 2 is operated at the stretch rated power level of 3083.4 MWt. These systems are listed in Tables 2 and 3. The review consisted of comparing the existing design criteria and stress analyses with the performance requirements at the power level of 3083.4 MWt and determining if modifications are required and if there is any impact on safety.

This report summarizes the results of the stretch rating review.

II. SUMMARY AND CONCLUSION

The BOP systems and components were reviewed to determine the impact of increasing the power level of Indian Point Unit 2 from the current licensed core power level of 2758 MWt to 3083.4 MWt NSSS. The review was performed to confirm that the BOP components and systems were designed for 3083.4 MWt and to determine that modifications made to the plant since its original licensing still maintain the units' capability for operation at the proposed stretch rated conditions.

Various systems as shown in Table 2 were evaluated and it was determined that the system parameters are satisfactory to envelope, with adequate margin, operation at the stretch rated power level of 3083.4 MWt and remain in compliance with currently applicable design criteria, industry codes and standards, and safety limits specified or assumed in the FSAR for BOP systems and equipment. The review demonstrated that no significant hardware modifications are anticipated in order to operate at the stretch rated NSSS power level of 3083.4 MWt.

The systems listed in Table 3 were deemed to operate independently of power level and therefore are not affected by the stretch rating.

III. BOP SYSTEM AND COMPONENT EVALUATION

Steam Turbine System

The Indian Point Unit 2 steam turbine system, originally guaranteed for flow conditions at 3083.4 MWt, was designed to be capable of a maximum steam flow rate of 13.94 x 10^6 lb/hr at an inlet steam pressure of 730 psia which corresponds to a thermal power rating of 3216 MWt. The original Low Pressure Turbine rotors were replaced in 1986 with a new forged rotor design. Also, the HP turbine was replaced in 1987 to provide greater reliability and improved performance. The replacement HP turbine rotor is operable (but not optimized) at the stretch rating conditions.

Main Feedwater System

The Main Feedwater System, a major system of the Indian Point Unit 2 Balance of Plant systems, is designed to supply a sufficient flow of feedwater to the four steam generators when they are operating at a thermal load of 3083.4 MWt (stretch rated). A review of the system design and principal component characteristics (including the two 60% capacity turbine driven main feedwater pumps) demonstrates that parameters such as pressure, temperature and flow, the characteristics of the flow control valves, and the heat transfer characteristics of the feedwater heaters as required for operation at 3083.4 MWt, are all bounded by the original design of the system. A review of the feedwater system is also discussed in the NSSS licensing report.

Main Steam and Reheat Steam System

The Main Steam System transports steam from the steam generators to the Main Turbine and Auxiliary Systems. A review of the system components confirms that the steam supply system requirements for operation at 3083.4 MWt are enveloped by the original design.

A review of the steam safety relief valves (discussed in the NSSS licensing report) confirms that they continue to meet the required relief capacities and technical specification criteria for the requested temperature range of operation (RCS Tavg of 549°F to 579.7°F), and at the stretch power level of 3083.4 MWt.

Main Moisture Separator/Reheater

The Moisture Separator/Reheater System provides dry low pressure steam to the low pressure turbines of the main turbine unit and to the main feed pump turbines. The major components of this system are the moisture preseparators, moisture separator/ reheaters, control valves, associated piping and instrumentation. Evaluation of equipment data sheets, pipe specifications, and system requirements demonstrates that the system is adequate to operate at the stretch rated power level of 3083.4 MWt. The moisture preseparators were not part of the original design, but were added to enhance system performance.

Main Turbine/Generator Auxiliary Systems

The following systems support the Main Turbine and Generator: Turbine/Generator Lube Oil, Main Turbine Control Oil, Generator H_2 Cooling, Generator Seal Oil, and Generator Stator Cooling. Each of these auxiliary systems has been confirmed as capable of supporting operation at the NSSS power level of 3083.4 MWt.

Steam Gland Sealing System

The function of the Steam Gland Sealing System is to prevent the leakage of air into, or steam from, the turbine cylinders. This system also serves the main boiler feed pump turbines.

At the maximum calculated power level (3216 MWt), which envelopes the stretch rated conditions of 3083.4 MWt, the main steam pressure will be within the gland steam control valve operating range. The Steam Gland Sealing System design bounds the requirements for operation at the stretch rated power level of 3083.4 MWt.

Lubricating Oil Storage, Transfer and Purification System

The Lubricating Oil Storage, Transfer and Purification System serves both the main turbine and the boiler feed pump turbines. The major components of this system are storage tanks, transfer pumps, purification and conditioning units, piping and instrumentation.

The main turbine and the boiler feed pump turbines were originally designed to envelope a power level of 3083.4 MWt. Recently a Lovejoy control system was installed to enhance the Main Boiler Feedwater Pump system reliability. The Lubricating Oil Storage, Transfer and Purification System also envelopes this power level. Since no additional load on the system is expected, the original design bounds operation at the stretch rated power level of 3083.4 MWt.

Extraction Steam, Heater Drains, and Vents System

The function of the Extraction Steam, Heater Drains, and Vents System is to take steam from the turbine cycle at various stages, condense it in the feedwater heaters, and return the water to the feed cycle through a cascading drain system. Evaluation of the equipment operating loads and the design parameters demonstrates that the stretch rated power level of 3083.4 MWt is bounded by the original plant design.

Condenser Air Removal System

Turbine efficiency increases with pressure drop across the turbine; therefore, by maintaining a design vacuum at the condenser, steam can perform more net work on the turbine. This vacuum is maintained by the Condenser Air Removal System. The system utilizes either steam jet air ejectors and/or mechanical vacuum pumps to remove non-condensible gases from the condenser. The Condenser Air Removal System was designed to support condenser operation at the stretch rated power level of 3083.4 MWt. In addition two supplementary vacuum pumps have been installed to provide operational flexibility.

Circulating Water System

The Circulating Water System utilizes six Circulating Water Pumps located in the intake structure to supply Hudson River water to the Main Condensers to condense the steam exhausting from the L.P. Turbines. A review of the Circulating Water System equipment and associated system requirements has confirmed that the system capacity is suitable for operation at the stretch rated power level of 3083.4 MWt.

Condensate System

The Condensate System delivers water from the main condenser hotwell to the suction of the steam generator feedwater pumps through five stages of low pressure feedwater heaters. The system consists of the main condensers, condensate pumps, low pressure feedwater heaters, and the associated piping and instrumentation. A review of the system equipment and the associated system requirements confirms that the system is adequate for operation at the stretch rated power level of 3083.4 MWt.

Condensate and Feedwater Chemical Feed System

The primary function of the Condensate and Feedwater Chemical Feed System is to store and inject chemicals into the Condensate and Feedwater systems to maintain proper water chemistry in the secondary piping and equipment. The Condensate and Feedwater Chemical System is capable of supporting operation of the Condensate and Feedwater Systems at 3083.4 MWt.

Boiler Feed Pump Lube Oil System

The Boiler Feed Pump Lube Oil System provides clean, water-free oil to the boiler feed pump and turbine drive bearings and to the BFP turbine control equipment.

The two Boiler Feed Pumps and their auxiliaries are each sized for 60% of full load operation (i.e., 3083.4 MWt). Therefore, the BFP Lube Oil System has the capability to operate safely at the stretch rating of 3083.4 MWt.

Steam Generator Blowdown System

The Steam Generator Blowdown System, in conjunction with the condensate and feedwater chemical feed system, maintains the secondary side water chemistry within NSSS specifications.

The Steam Generator Blowdown System was evaluated for a power level of 3083.4 MWt and a determination made that the existing blowdown system is adequate for the stretch rated conditions.

Primary Makeup Water System

The Primary Makeup Water System stores and supplies water for makeup to the reactor coolant system and other components that require demineralized water. The existing Primary Makeup Water System provides adequate capacity at the stretch rating of 3083.4 MWt.

Essential Service Water System

The Essential Service Water System continuously supplies the Hudson River cooling water to systems and components, such as containment fan cooler units and diesel generator coolers, which are necessary during normal and accident conditions. In addition, it supplies service water to the non-safety systems such as the turbine oil cooler.

The current containment integrity analyses, performed at a power level of 3216 MWt has reduced the cooling requirements for the containment fan cooler units (at 85°F) by approximately 20% resulting in added margin for the essential service water system at LOCA conditions. The existing service water system, as such, has adequate margin for supporting plant operation at the stretch rating of 3083.4 MWt.

Non-Essential Service Water System

The Non-Essential Service Water System supplies the Hudson River water to various heat loads in the secondary portions of the plant, including the component cooling heat exchanger (CCW) which does not require cooling immediately following a Design Basis Accident. The service water requirement for the CCW system during the recirculation phase of the LOCA are bounded under the current FSAR analysis.

During plant cooldown, the Residual Heat Removal System cools the Reactor Coolant from 350°F to 140°F. The heat load is transferred to the Component Cooling System and then to the Non-essential Service Water System. Due to the higher residual heat associated with the stretch rating of 3083.4 MWt, plant cooldown will require a somewhat longer, but acceptable, time without changing service water flow rates to the Component Cooling Heat Exchanger.

Other heat loads served by the Non-essential Service Water System have been reduced due to the lower flow requirements for certain components, such as the elimination of the Flash Evaporator. This results in greater margin in the Non-essential Service Water System for plant operation at the stretch rating of 3083.4 MWt.

Station Air/Instrument Air/Nitrogen

The Station Air System provides compressed air for pneumatic tools and miscellaneous cleaning and maintenance purposes throughout the plant. It also serves as an alternate supply to the Instrument Air System and as an emergency supply to the Containment Weld Channel and Penetration Pressurization system. The Instrument Air System provides clean, oil and moisture free air to the instruments, controls and other required services throughout the plant. Pneumatic valve actuators which require air pressure to go to their proper safeguards position during accident conditions have their Instrument Air Supply backed-up by a bottled Nitrogen gas supply.

The operation of the plant at the stretch rating of 3083.4 MWt will not produce any change in the Station Air, Instrument Air, or N_2 systems.

Secondary Sampling System

The Secondary Sampling System provides representative samples of non-nuclear process fluids for analyses necessary for plant operation, corrosion control, and monitoring of equipment and system performance.

No additional sampling points will be required for operation at the stretch rated power level of 3083.4 MWt. Slight changes in various fluid parameters at the stretch rated conditions are within the sampling capability of the present system.

Solid Radwaste System

The Solid Radwaste System collects and processes solid waste prior to storage or disposal. It is unlikely that stretch rated operation will increase system demand. However, this is a batch process and any increase in system processing requirements due to plant operation at the stretch rating of 3083.4 MWt would be accommodated by more frequent batch operations.

Containment Design

The Containment Building completely encloses the Reactor and Reactor Coolant System main loop piping and ensures that essentially no leakage of radioactive material would result even if gross failure of the Reactor Coolant System were to occur. The design pressure for the containment structure is 47 psig, which is in excess of calculated peak pressures created by the hypothetical loss-of-coolant accident.

The latest Containment Integrity Analysis, performed at a power level of 3216 MWt, calculates a peak containment pressure of 40.5 psig. Therefore, the Containment design appropriately bounds calculated accident conditions at the stretch rating of 3083.4 MWt.

Containment Integrated Leak Rate Test (ILRT)

The Integrated Leak Rate Test is performed with the containment pressurized to a minimum of 47 psig (the design pressure) for a period of 24 hours. The test verifies that the containment leakage rate is no greater than 0.1 percent by weight of the containment volume per day at the Design Basis Accident conditions.

The latest Containment Integrity Analysis, performed at a power level of 3216 MWt, calculates a peak containment pressure of 40.5 psig. Therefore, the containment design appropriately bounds the calculated accident conditions, and the ILRT conditions are unaffected by plant operations at the stretch rating of 3083.4 MWt.

Main Control Room Panels

The Main Control Room Panels provide a centralized control facility for the plant operators to use for monitoring and controlling the plant operation. Functions such as starting, stopping, tripping, and control of major plant equipment are accomplished from the Main Control Room Panels.

Since no additional equipment requiring control from the Main Control Room is required for stretch rated power level operation of 3083.4 MWt, the Main Control Room Panels are not affected by stretch rating.

Miscellaneous Control Panels

The Miscellaneous Control Panels provide control facilities for equipment and systems generally requiring less attention than those controlled from the main control panel. Functions such as starting, stopping, monitoring, and control of equipment are accomplished from the Miscellaneous Control Panels.

Since no additional equipment requiring control from the Miscellaneous Control Panels will be required for stretch rated power level (3083.4 MWt) operation, the Miscellaneous Control Panels are not affected by stretch rating.

Plant Annunciator

The Plant Annunciator System provides visual and audible alarms to alert the operator to an off normal condition in the plant.

Since no additional equipment requiring annunciation will be required for stretch rated power level operation of 3083.4 MWt, the Plant Annunciator System is not affected by stretch rating.

Heating Ventilating & Air Conditioning System

The following systems were reviewed:

- Control Room HVAC
- Fuel Storage Building HVAC
- Containment Purge & Pressure Relief

It is concluded that there will be no impact on these systems due to plant operations at the stretch rating of 3083.4 MWt.

Main Generator

The Main Generator generates electrical power at 22 kV, which is then transmitted through an isolated phase bus and two half-sized 22/345 kV main power transformers to the offsite power system. The generator is also the main source of 6.9 kV auxiliary electrical power during normal plant operation. Power to the auxiliaries is supplied by a 22/6.9 kV two winding unit auxiliary transformer that is connected to the isolated phase bus from the generator.

The Main Generator is rated at approximately 1300 MWe. Therefore, the Main Generator has adequate capacity for stretch rated conditions corresponding to approximately 1022 MWe.

Main Transformer

The main generator feeds electrical power at 22 kV through an isolated phase bus to two half-sized 22/345 kV Main Power Transformers. The transformers, each rated at 607,040 kVA at 65°C average rise, are adequate for the operation at stretch rated conditions.

Outgoing Extra High Voltage (EHV) Lines

Outgoing EHV lines deliver the output of the station to the Buchanan Substation at 345 kV.

The EHV lines are adequately sized for the stretch power level of 1022 MWe; no impact on the outgoing EHV lines is expected.

Start-Up & Unit Auxiliary Transformer

The Station Auxiliary Transformer (Start-Up Transformer) and the Unit Auxiliary Transformer are each rated at 48,160 kVA at 65°C while being supplied by forced oil and forced air cooling. These transformers are adequately sized for existing loads. Stretch rated (3083.4 MWt/1022 MWe) operation does not require any additional equipment, or operation of the existing equipment beyond present loading for NSSS Systems.

Further, no additional equipment is required for stretch rated condition for BOP systems. Some of the Balance of Plant loads, such as Condensate pumps, will be loaded more than their present loading. However, the new loading will not exceed the nameplate rating. Therefore, no significant impact on the Start-up and Unit Auxiliary Transformers is expected.

6.9 kV AC System

The 6.9 kV AC System supplies power to various loads in the Balance of Plant as well as NSSS Systems. These systems are required for normal or emergency mode of operation.

The 6.9 kV AC System is adequately sized for existing loads. Stretch rated (3083.4 MWt/1022 MWe) condition does not require any additional 6.9 kV AC equipment or operation of the existing equipment beyond their present loading for NSSS Systems. Therefore, no impact on the 6.9 kV AC system is expected.

480 V AC System

The 480 V AC System supplies various 480 V AC loads in Balance of Plant as well as NSSS systems. These systems are required for normal operation and safe shutdown.

The 480 V AC System is adequately sized for existing loads. Stretch rated operation (3083.4 MWt) does not require additional equipment or operation of the existing equipment beyond their original loading for NSSS Systems. Therefore, no impact on the 480 V AC system is expected.

120 V Instrument AC System

The 120 V Instrument AC system provides power for the instrumentation and control of the reactor and turbine generator utilizing four instrument buses. All four instrument buses have a source of power from the 125 V DC buses through static inverters (one for each bus). Each inverter converts the DC voltage into a 118 Volt single phase 60 Hertz regulated power supply.

The 120 V Instrument AC system is sized for full load equipment operation. Since operation at the stretch rating does not require any additional 120 V AC equipment, or operation of the existing equipment beyond present loading, no impact on the 120 V AC system is expected.

125 V DC System

The DC System is provided for control, protection, emergency lighting and a supply for the static inverters which supply the four instrument buses. The system is comprised of four lead acid batteries, four power panels and four distribution panels. Each power panel connects to a battery charger that carries the normal DC load while maintaining the batteries fully charged. The system is required for normal operation and safe shutdown.

The 125 V DC system has been evaluated at the stretch rating of 3083.4 MWt (1022 MWe) and it was determined that the system was not impacted under the stretch rated operation.

Emergency Diesel Generators

The Emergency Diesel Generators function as a backup to the normal AC power supply and provides the power required for safe shutdown of the reactor in the event of a loss of the preferred source of power. Any two units are capable of sequentially starting and supplying the power requirement of one complete set of safeguards equipment.

The loads supplied by the Emergency Diesel Generators were examined and they continue to remain at the existing values.

Emergency Fuel Oil Supply

The function of the Emergency Fuel Oil Supply System is to store and supply sufficient fuel oil to the Emergency Diesel Generator sets for operation of the minimum required engineered safeguards for a period of 168 hrs (7 days).

Since the emergency loads to the emergency diesel generator are not affected by operating the plant at the stretch rated (3083.4 MWt) conditions, no impact on the Emergency Fuel Supply System is expected.

Load Shedding and Emergency Load Sequencing

The Load Shedding and Emergency Load Sequencing System provides for removal of selected loads from Class 1E buses in the event of blackout plus unit trip, or upon initiation of safety injection signals, and reloads the equipment in a predetermined sequence. This ensures that voltage and frequency of the Class 1E buses is not degraded due to heavy starting of actuated equipment.

Since no new or increased loads are being added to the emergency buses, the load shedding and subsequent reloading of equipment on Class 1E buses will be unaffected by operating at the stretch rating of 1022 MWe.

TABLE 2

SYSTEMS EVALUATED FOR POTENTIAL IMPACT DUE TO STRETCH RATING

DESCRIPTION

Steam Turbine System
Main Feedwater System
Main Steam and Reheat Steam System
Secondary/Auxiliary Piping (Refer to Section IV for narrative)
Main Moisture Separator/Reheater
Main Turbine/Generator Auxiliary Systems
Steam Gland Sealing System
Lubricating Oil Storage, Transfer & Purification System
Extraction Steam, Heater Drains & Vents System
Condenser Air Removal System
Circulating Water System
Condensate System
Condensate and Feedwater Chemical Feed System
Boiler Feed Pump Lube Oil System
Steam Generator Blowdown System
Primary Makeup Water System
Essential Service Water System
Non-Essential Service Water System
Station Air/Instrument Air/Nitrogen (N_2)
Secondary Sampling System
Solid Radwaste System
Containment Design
Containment Integrated Leak Rate Test (ILRT)
Main Control Room Panels
Miscellaneous Control Panels
Plant Annunciator
Heating Ventilating & Air Conditioning System
Main Generator
Main Transformer
Outgoing Extra High Voltage (EHV) Lines
Startup & Unit Auxiliary Transformer
6.9 kV AC System
480 V AC System
120 V Instrument AC System
125 V DC System
Emergency Diesel Generators
Emergency Fuel Oil Supply
Load Shedding & Emergency Load Sequencing

IV. PIPING SYSTEMS

Secondary/Auxiliary Piping

A detailed review of original pipe stress analyses and technical evaluation was performed for the BOP piping. This review/evaluation confirmed that the pipe stresses for the range of temperatures and pressures at stretch rated conditions remain within the applicable code allowable limits.

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

V. MISCELLANEOUS ISSUES

Equipment Qualification

Evaluation of the equipment qualification at the stretch rated power level has been performed. The review/analysis confirmed that the equipment is qualified for temperature, pressure and radiation levels corresponding to operation at 3083.4 MWt.

Turbine Missile Analysis

The LP turbine rotors replaced in 1986 with a fully integral design provide improved protection against any postulated turbine missile event as these rotors do not incorporate the original "keyway/disk" design. Further, there is no increase in the speed of the turbines as a result of operation at higher power levels. Therefore, the turbine missile analysis is not affected by operation under stretch rating conditions.

Flooding

A review of the existing analyses on the susceptibility of safety-related systems to flooding caused by the failure of Non-Category I systems was performed. It showed that the potential sources of flooding are controlled by systems which are not affected by the stretch rated conditions.

Fire Hazards

The fire protection systems are independent of the power level and thus will not be affected by the stretch rating.

VI. SYSTEMS NOT DEPENDENT UPON THERMAL POWER LEVEL

Consideration was given to all systems with respect to possible effects due to stretch rating operation. It was determined that many systems are not dependent upon the thermal power level and would therefore not require detailed review. These systems are listed in Table 3 entitled "Systems Not Dependent Upon Thermal Power Level."

As part of this effort two typical systems were selected for a limited review. Short narratives for these two typical systems are as follows:

Secondary Liquid Waste Storage (Blowdown Purification System)

This system diverts the blowdown stream from the Unit 2 Blowdown Flash Tank to the Support Facilities Waste Collection System in the event of primary to secondary leakage in one or more of the steam generators. Since the system design is independent of power level, it is unaffected by the increase in power to the stretch rating of 3083.4 MWt.

Plant Steam System

The IP-2 plant steam system, as an extension to the Unit 1 heating system, supplies low pressure steam for heating throughout the plant and for maintenance purposes. The system is independent of plant power output.

TABLE 3

SYSTEMS NOT DEPENDENT UPON THERMAL POWER LEVEL

DESCRIPTION

Secondary Liquid Waste Storage (Blowdown Purification)*
Hydrogen (H_2), Oxygen (O_2), and Carbon Dioxide (CO_2) Gas
Service Water Chlorination System
House Service Boiler
Plant Steam System*
Turbine/Heater Bay Building HVAC
Miscellaneous Building HVAC
PAB & Tank Pit Building HVAC
Auxiliary Feedwater Building HVAC
Electric Penetration Tunnel HVAC
Diesel Generator Building HVAC
Decontamination
Breathing Air (Control Room)
Fire Protection
Fuel Handling Storage & RV Services
Cranes, Hoists & Elevators
Bulk Chemical Storage
Sanitary Drainage
Roof and Yard Drains
Oil Waste
Floor and Equipment Drains
Normal Lighting
Emergency Lighting DC
Telephone System
Public Address
Grounding
Cathodic Protection
Freeze Protection
Safety Assessment System
Process Radiation Monitoring
Area Radiation Monitoring
Meteorological Instrumentation
Seismic Instrumentation
Plant Security
Emergency Response Facility Information
Radioactive Release Information

* Limited writeups are attached confirming no effect due to an increase in power level. For the other systems, operation is not influenced by increased power level and no writeups are attached.

ENCLOSURE 3

to

ATTACHMENT B

No Significant Hazards Consideration Evaluations

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
September 30, 1988

1.0 NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION FOR REQUESTED INCREASE IN AUTHORIZED RATED THERMAL POWER

1.1 Description of Change

The proposed revisions to Indian Point Unit 2 Facility Operating License, Section 2.C.1 and Technical Specification 1.1.a, require changes to indicate operation is authorized at core power levels not in excess of 3071.4 MWt.

1.2 Safety Assessment

The plant was originally designed to operate at a power level equal to or greater than that sought by the present application. In support of the presently requested increase in authorized core power level to 3071.4 MWt (3083.4 MWt, NSSS), the limiting aspects of the NSSS system and component design were analyzed to verify that at the requested stretch power level, there would be compliance with licensing criteria and standards currently required by the Indian Point Unit 2 operating license, as well as the functional requirements specified in the FSAR. The review was conducted in accordance with the methodology documented in Westinghouse topical report WCAP-10263, entitled "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant." This methodology has been referenced in connection with the approval of similar license amendments for other facilities. WCAP-10263 methodologies were also employed in the review of the interfaces between the NSSS and the Balance of Plant (BOP) systems and components. Reviews of the BOP systems and components were conducted and their capability to meet the requirements for operation at 3083.4 MWt. verified. These reviews indicate that the Indian Point Unit 2 plant will be capable, in its present design configuration, of operating at the stretch power level without violating any of the design or safety limits specified in the Indian Point Unit 2 FSAR or Facility Operating License.

Con Edison has submitted a license amendment request, dated September

30,1988, to utilize Westinghouse Optimized Fuel Assemblies (OFA) beginning with Cycle 10 operation. In conjunction with the use of OFA, accident analyses were performed at the stretch rated power conditions. The results of the accident analyses indicate that all applicable safety limits will be met at the stretch rated conditions. Confirming analysis of the non-limiting evaluations will be submitted as needed to support this amendment request.

1.3 Basis for No Significant Hazards Consideration Determination

It is expected that upon completion of all analysis the results will show that the proposed changes do not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. This is based on the original design basis of the plant, as confirmed by the analyses supporting original plant licensure. These include an environmental evaluation which assumed stretch rated conditions and a radiological evaluation conducted at 3216 MWt. These analyses have been further confirmed by analyses performed persuant to the methodology of WCAP-10263, as described above.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. This is based on the system and components reviews accompanying original plant licensure, as confirmed by analyses recently conducted, all of which verify the capability of systems and components to operate at the stretch rated conditions.

(3) involve a significant reduction in a margin of safety. Accident analyses, both past and present, performed at the stretch rated conditions demonstrate that DNB design basis remains unchanged, that the RCS pressure limit of 2700 psig will not be exceeded,

and that LOCA results remain well below the regulatory limits given in 10 CFR 50.46.

Based on the above discussions, it is concluded that the amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated; does not create the possibility of a new or different kind of accident from any accident previously evaluated; and does not involve a reduction in a required margin of safety. Therefore, the requested amendment does not present any significant hazards considerations.

2.0 NO SIGNIFICANT HAZARDS CONSIDERATION FOR OTDT and OPDT Setpoints

2.1 Description of Change

The proposed revision to Technical Specifications 2.3.1.B(4) and 2.3.1.B(5) consists of changes to the nominal average temperature value at rated power for the Overtemperature Delta T and Overpower Delta T protection logic functions.

The proposed changes are necessary to reflect the increased average temperature allowed at the increased power level.

2.2 Safety Assessment

The proposed revisions to Technical Specification 2.3.1.B(4) and 2.3.1.B(5) would revise the Delta T limit at which a reactor trip signal would be produced as a result of increases in Delta T from either overtemperature or overpower transient conditions. Such a revision does not limit operation flexibility nor affect the safety function of the reactor trip signals on the Delta T protection. These revisions only change the allowable Delta T limit functions to be consistent with plant protection required to bound the reactor core safety limits specified for Technical Specification Figure 2.1-1.

All of the licensing basis accidents described in FSAR Chapter 14 which take credit for an OTDT or OPDT reactor trip will be or have been analyzed considering the proposed changes to the OTDT and OPDT setpoint functions as provided in the proposed changes to Technical Specification 2.3.1.B(4) and (5). The results of analyses performed to date have demonstrated conformance with the applicable design and regulatory requirements assuming the proposed change for reactor trip on the Delta T protection setpoint functions. Those analyses remaining to be perfomed are expected to also demonstrate continued comformance with the applicable design and regulatory requirements.

2.3 Basis for No Significant Hazards Consideration Determination

It is expected that upon completion of all analyses the results will show that the proposed changes do not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision is being supported by conservative evaluation and analyses utilizing the latest approved computer codes and methodology. These analyses are expected to demonstrate conformance to the applicable design and regulatory criteria.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes to the OTDT and OPDT setpoint functions for reactor trip do not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require evaluation and resolution. Nothing would be added or removed that would conceivably introduce a new or different kind of accident mechanism or initiating circumstance than that previously evaluated.

In general, the proposed changes do not adversely effect the ability of OTDT and OPDT reactor trip signals to perform their safety function to initiate reactor core shutdown during an overtemperature Delta T or overpower Delta T transient condition, respectively.

(3) involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are expected to be met, remain conservative, and continue to maintain the previous margins of safety.

The safety function of reactor trip on overtemperature delta T and overpower delta T is to initiate reactor core shutdown during delta T transient events to ensure that the reactor core safety limits as defined in Technical Specification Figure 2.1-1 are not exceeded. Safety evaluations and analyses for all of the licensing basis accidents described in FSAR Chapter 14 which take credit for an OTDT or OPDT reactor trip have been or will be performed and the results of these analyses are expected to demonstrate conformance with the applicable design and regulatory requirements.

Therefore, based on the above, we conclude that the proposed changes are not expected to constitute a significant hazards consideration.

3.0 NO SIGNIFICANT HAZARDS CONSIDERATION FOR DNB Parameters

3.1 Description of Change

The proposed change to Technical Specification 3.1.G would increase the allowable Tavg to 585.5 F. The proposed change is necessary to reflect the increased average temperature allowed at the increased power level.

3.2 Safety Assessment

The safety analyses and evaluations performed in support of the rated power increase consider the increase in Tavg at 100% power. The results of analyses performed to date have demonstrated conformance with the applicable design and regulatory requirements assuming the proposed change to average temperature. Those analyses remaining to be perfomed are expected to also demonstrate continued comformance with the applicable design and regulatory requirements.

3.3 Basis for No Significant Hazards Consideration Determination

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (51 FR 7751). Example (vi) of those involving no significant hazards consideration discusses a change which may reduce a safety margin but where the results are clearly within all acceptance criteria with respect to the system or component. The proposed change is to increase the allowable Reactor Coolant System average temperature at 100% power.

It is expected that upon completion of all analyses the results will show that the proposed changes do not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The Tavg value represents a design limit for average Reactor Coolant System temperature. This proposed change will be supported by conservative analyses and evaluations based on approved codes and methodologies. All applicable design and safety criteria continue to be satisfied.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change in the design value of Tavg does not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require evaluation or resolution. Nothing would be added or removed that would conceivably introduce a new or different kind of accident mechanism or initiating circumstances than that previously evaluated.

(3) involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are expected to be met, remain conservative, and continue to maintain the previous margins of safety. Approved analysis codes and methodologies are being employed as the basis for evaluating this proposed change.

All applicable design and safety criteria are expected to be satisfied including the impact of an increased Tavg.

Therefore, based on the above, we conclude that the proposed changes are not expected to constitute a significant hazards consideration.

4.0 NO SIGNIFICANT HAZARDS CONSIDERATION FOR
Auxiliary Feedwater Flowrate

4.1 Description of Change

The proposed revision to Technical Specification 3.4.A.2 would maintain the performance capability and reliability of the Steam and Power Conversion System.

4.2 Safety Assessment

The proposed revision to Technical Specification 3.4.A.2 would revise the minimum flow capability of each of the auxiliary feedwater pumps. Such a revision does not limit operation flexibility nor affect the safety function of the auxiliary feedwater pumps. Under the proposed amendment, this Technical Specification would be revised to maintain the minimum flow capability of each of the three auxiliary feedwater pumps at 380 gpm.

The safety function of the auxiliary feedwater system is to supply high-pressure feedwater to the steam generators to maintain a water inventory. This is needed to remove decay heat energy from the reactor coolant system by secondary side steam release in the event that the main feedwater system is inoperable. The most severe of these events are the Loss of Normal Feedwater and Loss of all AC power to the station auxiliaries. These events have been reanalyzed using the revised minimum auxiliary feedwater flow rate. The results of these analyses have clearly demonstrated conformance with the applicable design and regulatory requirements.

4.3 Basis for No Significant Hazards Consideration Determination

Consistent with the Commission's criteria in 10 CFR 50.92, we have determined that the proposed change does not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision is supported by conservative evaluation and analyses utilizing the latest approved computer codes and methodology. These analyses have demonstrated conformance to the applicable design and regulatory criteria.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change to the minimum auxiliary feedwater pump flowrate does not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require evaluation and resolution. Nothing would be added or removed that would conceivably introduce a new or different kind of accident mechanism or initiating circumstance than that previously evaluated.

In general, the proposed change does not adversely effect the ability of the auxiliary feedwater system to perform its safety function to supply high pressure feedwater to the steam generators to maintain a water inventory.

(3) involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety.

The safety function of the auxiliary feedwater system is to supply high pressure feedwater to the steam generators to maintain a water inventory. Safety evaluation and analyses for all of the licensing basis accidents described in FSAR Chapter 14 which take credit for the auxiliary feedwater system have been performed and the results of these analyses and evaluation have demonstrated conformance with the applicable design and regulatory requirements.

Therefore, based on the above, we conclude that the proposed change does not constitute a significant hazards consideration.

5.0 NO SIGNIFICANT HAZARDS CONSIDERATION FOR Secondary Side Steam Flow

5.1 Description of Change

The proposed technical specification changes (Section 3.4 Basis) are administrative in nature and are made to reflect the increased steam flow and power level resulting from the proposed increase in licensed NSSS power. The total secondary side steam flow value is changed to 13,310,000 lbs/hr to reflect the steam flow at a power level of 3083.4 Mwt. The percentage of total main steam safety valve relieving capacity that this steam flow represents is also changed as a result of the increase in steam flow.

5.2 Safety Assessment

The proposed changes are purely administrative changes to reflect values which change as a result of the proposed increase in licensed NSSS power. The proposed changes have no effect on the safety function of any systems, components, or operation. The proposed changes have no effect on the application of any specifications, and is therefore purely an administrative change. No safety analyses are affected by the proposed changes.

5.3 Basis For No Significant Hazards Consideration Determination

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (51 FR 7751) of amendments that are considered no likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

The proposed changes are purely administrative changes to achieve consistency with the Technical Specifications, and consistency with the proposed increase in licensed NSSS power.

Consistent with the Commission's criteria in 10CFR50.92, we have determined that the proposed changes described above do not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision do not affect plant operations. The proposed revisions provide consistency with Technical Specifications associated with the proposed increase in licensed NSSS power.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not modify the plant's configuration or operation. Nothing would be added or removed that would conceivably introduce a new or different kind of accident mechanism or initiating circumstance that those previously evaluated.

(3) involve a significant reduction in a margin of safety. With the proposed changes, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety. Because these changes are administrative in nature their implementation does not affect any margin of safety.

Therefore, based on the above, we conclude that the proposed change does not constitute a significant hazards consideration.