October 22, 1996

Mr. William J. Cahill, Jr.
Chief Nuclear Officer
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M96086)

Dear Mr. Cahill:

The Commission has issued the enclosed Amendment No. 170 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3 (IP3). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated March 29, 1996, as supplemented July 12, 1996, and September 6, 1996.

The proposed amendment would change the TSs relating to minimum reactor coolant system flow and maximum RCS average temperature to make these parameters consistent with an assumption of 100% helium release from the boron coating of the integral fuel burnable absorber rods. The proposed amendment would also add limits associated with Departure from Nucleate Boiling to the IP3 TSs.

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

Sincerely,

/S/

George F. Wunder, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures: 1. Amendment No. 170 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

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ISSUANCE OF AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. DPR-28

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G. Hill (2)
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 22, 1996

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cc w/encls: See next page

William J. Cahill, Jr.
Power Authority of the State
of New York

Indian Point Nuclear Generating
Station Unit No. 3

cc:

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Gerald C. Goldstein
Assistant General Counsel
Power Authority of the State
of New York
1633 Broadway
New York, NY 10019

Mr. Robert G. Schoenberger
President and Chief Operating
Officer
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

Mr. Robert J. Barrett
Plant Manager
Indian Point 3 Nuclear Power Plant
P.O. Box 215
Buchanan, NY 10511

Ms. Charlene D. Faison
Director Nuclear Licensing
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

Mr. F. William Valentino, President
New York State Energy, Research,
and Development Authority
2 Rockefeller Plaza
Albany, NY 12223-1253

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Resident Inspector
Indian Point 3 Nuclear Power Plant
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, NY 10511

Mr. Charles W. Jackson
Manager, Nuclear Safety and
Licensing
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenues
Buchanan, NY 10511

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Mr. Richard L. Patch, Director
Quality Assurance
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

Union of Concerned Scientists
1616 P Street, NW, Suite 310
Washington, DC 20036

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 170
License No. DPR-64

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated March 29, 1996, as supplemented July 12, 1996, and September 6, 1996, 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 changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-64 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 170 , 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

S. Singh Bajwa, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 22, 1996

ATTACHMENT TO LICENSE AMENDMENT NO.170

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
i	i
ii	ii
iv	iv
1-1	1-1
2.1-1	2.1-1
2.1-2	2.1-2
Figure 2.1-1	Figure2.1-1
3.1-4	3.1-4
3.1-8	3.1-8
3.1-36	3.1-36
3.1-37	3.1-37
3.1-38	3.1-38
3.1-39	3.1-39
Table 4.1-1 (Sheet 1 of 6)	Table 4.1-1 (Sheet 1 of 6)
Table 4.1-1 (Sheet 6 of 6)	Table 4.1-1 (Sheet 6 of 6)
4.3-1	4.3-1
4.3-4	4.3-4

APPENDIX A - TECHNICAL SPECIFICATIONS AND BASES
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

TABLE OF CONTENTS

Amendment No. 90, 101, 115, 116, 170

Amendment No. 83, 94, 101, 115, 116, 121, 170

Amendment No. 30, 59, 88, 103, 116, 170

TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

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

1.1 REACTOR CONDITIONS

1.1.1 Rated Thermal Power (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3025 MWt. ("Rated Power" and "Rated Thermal Power" are used interchangeably throughout the Technical Specifications).

1.1.2 Thermal Power

Thermal Power shall be the total reactor core heat transfer rate to the reactor coolant.

1.1.3 Reactor Pressure

The pressure in the steam space of the pressurizer.

1.1.4 Tavg

Average temperature across the reactor vessel as measured by the hot and cold leg temperature detectors.

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 but $\leq$ 555°F.

2.0 Safety Limits and Limiting Safety System Settings

2.1 Safety Limits, Reactor Core

Applicability

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

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1 for four-loop operation. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System vessel inlet temperature and power level is at any time above the appropriate pressure line.

Basis

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature. The safety limits represent a design requirement for establishing the trip setpoints identified in Technical Specification 2.3. Technical Specification 3.1.H, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," provide more restrictive limits to ensure that the safety limits are not exceeded.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore thermal power and Reactor Coolant Temperature and Pressure have been related to DNB through the WRB-1 correlation for Westinghouse Optimized fuel. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: There must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I (normal operation and operational transients) and Condition II (events of moderate frequency) events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95% probability with 95% confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability with 95% confidence level that the minimum DNBR for the limiting rod is greater than or equal to the applicable DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. The DNBR uncertainty combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin is maintained by performing DNB design evaluations to a higher DNBR value, called the Safety Limit DNBR.

The curves of Figure 2.1-1 show the loci of points of thermal power, Reactor Coolant System pressure and vessel inlet temperature for which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The calculation of these limits includes:

1. $F_{\Delta H}^{RTP} = F_{\Delta H}^{N}$ limit at Rated Thermal Power (RTP) specified in the COLR.

2. an equivalent steam generator tube plugging level of up to 30% in any steam generator provided the equivalent average plugging level in all steam generators is less than or equal to 24%,

3. a reactor coolant system total flow rate of greater than or equal to 385,400 gpm as measured at the plant,

4. a reference cosine with a peak of 1.55 for axial power shape.

Figure 2.1-1 includes an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$$F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP} \left(1 + PF_{\Delta H} \left(1-P\right)\right)$$

Where P is the fraction of Rated Thermal Power.

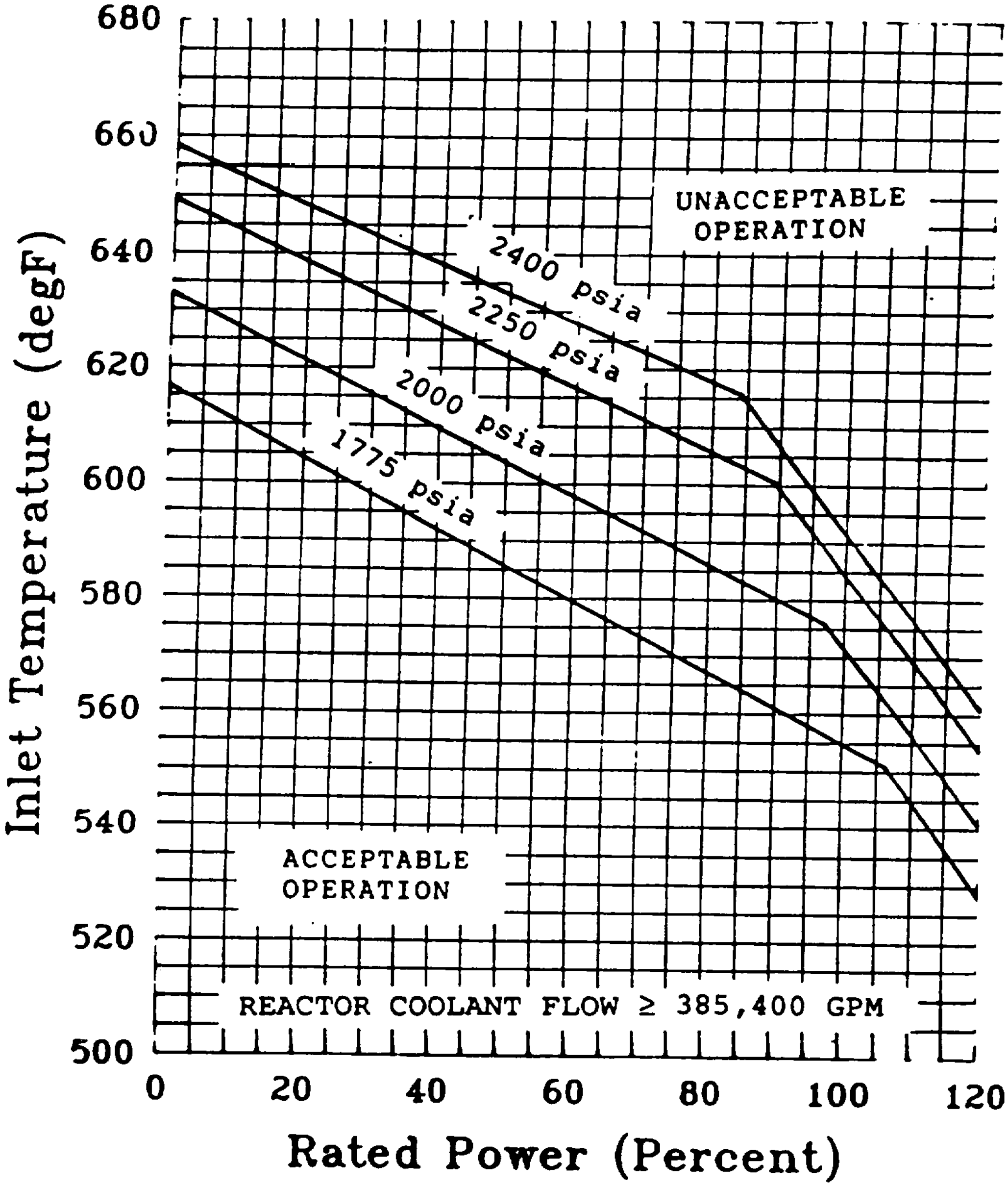
$F_{\Delta H}^{RTP}$ is the $F_{\Delta H}^{N}$ limit at Rated Thermal Power specified in the COLR, and $PF_{\Delta H}$ is the Power Factor Multiplier specified in the COLR.

When flow or $F_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the limits presented. A 2.6% measurement uncertainty on Flow and a 4% measurement uncertainty of $F_{\Delta H}$ have already been included in the above limits.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit (specified in the COLR) assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

REACTOR CORE SAFETY LIMITS

This curve does not provide allowable limits for normal operation.
(See Technical Specification 3.1.H for DNB limits)



100 PERCENT RATED POWER IS EQUIVALENT TO 3025 MWt

Pressures and temperatures do not include allowance for instrument error.

FIGURE 2.1-1

Amendment No. ~~86~~, 170

2. Safety Valves

a. At least one pressurizer code safety valve shall be operable, or an opening greater than or equal to the size of one code safety valve flange to allow for pressure relief, whenever the reactor head is on the vessel except for hydrostatically testing the RCS in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is above the cold shutdown condition except during reactor coolant system hydrostatic tests and/or safety valve settings.

c. The pressurizer code safety valve lift setting shall be set at 2485 psig with ±1% allowance for error.

3. Pressurizer Heaters

Whenever the reactor is above the hot shutdown condition, the pressurizer shall be operable with at least 150 kw of pressurizer heaters.

a. With less than 150 kw of pressurizer heaters operable, restore the required inoperable heaters within 72 hours or be in at least hot shutdown within an additional 6 hours.

4. Power Operated Relief Valves

Whenever the reactor coolant system is above 400°F, the power operated relief valves (PORVs) shall be operable or their associated block valves closed.

a. If the block valve is closed because of an inoperable PORV, the control power for the block valve must be removed.

b. If the above conditions cannot be satisfied within 1 hour, be in at least hot shutdown within 6 hours and in cold shutdown within the following 30 hours.

5. Power Operated Relief Block Valves

Whenever the reactor coolant system is above 400°F, the motor operated block valves shall be operable or closed.

a. If the block valve is inoperable, the control power is to be removed.

b. If the above conditions cannot be satisfied within 1 hour be in at least hot shutdown within the following 30 hours.

6. Deleted

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal off possible RCS leakage paths.

Reactor vessel head vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor vessel head vent path ensures that capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

The OPS is designed to relieve the RCS pressure for certain unlikely incidents to prevent the peak RCS pressure from exceeding the 10 CFR 50, Appendix G, limits. "Arming" means that the motor operated valve (MOV) is in the open position. This can be accomplished either automatically by the OPS when the RCS temperature is less than or equal to 332°F or manually by the control room operator.

3.1 Reactor Coolant System (RCS)

H. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

Specification

1. During the POWER OPERATION CONDITION, RCS DNB parameters for pressurizer pressure and RCS average temperature shall be within the limits specified below:

   a. Pressurizer pressure ≥ 2205 psig;

   b. Maximum indicated T_{avg} ≤ 571.5°F; and

2. At the POWER OPERATION CONDITION with four reactor coolant pumps running, the RCS DNB parameter for RCS total flow rate shall be within the following limit:

   RCS total flow rate ≥ 385,400 gpm.

3. The pressurizer pressure limit of Specification 3.1.H.1 does not apply during:

   a. THERMAL POWER ramp > 5% RTP per minute; or

   b. THERMAL POWER step > 10% RTP.

4. If pressurizer pressure, RCS average temperature, or RCS total flow rate are not in accordance with Specifications 3.1.H.1, 3.1.H.2, or 3.1.H.3, then, immediately verify that the safety limits of Specification 2.1 have not been exceeded and, within 2 hours, restore the RCS DNB parameter(s) to within limits.

5. If pressurizer pressure and/or RCS average temperature are not restored to within limits within 2 hours, be in the HOT SHUTDOWN CONDITION within 6 hours.

6. If RCS total flow rate is not restored to within the limits of Specification 3.1.H.2 within 2 hours, bring THERMAL POWER to ≤ 10% RTP within 6 hours and ensure operation is in accordance with Specification 3.1.A.1.e.

Surveillance Requirements

Reference Technical Specification Table 4.1-1, Items 4, 5, and 7, and Section 4.3.B.

Bases

Background

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure and temperature limits are consistent with operation within the nominal operational envelope. A lower pressure will cause the reactor core to approach DNB limits. A higher RCS average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit bounds that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

Applicable Safety Analyses
The requirements of this Specification represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this Specification will result in meeting the applicable DNBR criteria. Changes to the unit that could affect these parameters must be assessed for their effect on the DNBR criteria.

Specification
Specifications 3.1.H.1 and 3.1.H.2 specify limits on the monitored process variables (pressurizer pressure, RCS average temperature, and RCS total flow rate) to ensure that the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

The RCS total flow rate limit of 385,400 gpm allows for a measurement uncertainty of 2.6% associated with the performance of Reactor Coolant System Flow Calculation required by Technical Specification 4.3.B. Because the flow instrumentation provides flow indication based on a percentage of full flow, the 385,400 gpm is converted into a percentage of full flow to accomodate the verification that RCS total flow is within limits during channel checks.

The pressurizer pressure limit of 2205 psig allows for measurement uncertainty and instrument error. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit.

The limit on maximum indicated RCS average temperature provides assurance that RCS temperatures are maintained within the normal steady state envelope of operation assumed in the safety analyses performed to support the Vantage 5 fuel reloads with asymmetric tube plugging among steam generators. A maximum full power T_{avg} of 547.9°F (including control deadband and measurement uncertainties) was assumed in these safety analyses. A T_{avg} of 578.3 F assures that a T_{avg} of 547.9°F is not exceeded at a measured flow of ≥ 385,400 gpm when considering asymmetric tube plugging among steam generators for DNB considerations. However, T_{avg} will be controlled to a maximum indicated T_{avg} of 571.5°F which assures consistency with analyses for post-LOCA containment integrity.

Applicability
During the POWER OPERATION CONDITION, the limits on pressurizer pressure and RCS coolant average temperature must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. For the same reason, during the POWER OPERATION CONDITION with four reactor coolant pumps running, the limit on RCS flow rate must be maintained. In all other operating conditions, the power level is low enough that DNB is not a concern.

Specification 3.1.H.3 indicates that the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counter productive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in Safety Limit 2.1, "Safety Limits, Reactor Core." Those limits are less restrictive than the limits of this specification but violation of a Safety Limit merits stricter, more severe required action. Should a violation of Specification 3.1.H.1 occur, the operator must check whether or not a Safety Limit has been exceeded.

Actions
RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within specification limits, action must be taken to restore the parameter(s).

The 2 hour completion time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience for Westinghouse plants.

If the required action of Specification 3.1.H.4 is not met within the associated completion time, the plant must be brought to a mode in which Specification 3.1.H.1 does not apply. To achieve this status, the plant must be brougnt to at least the HOT SHUTDOWN CONDITION within 6 hours. The reduced power condition eliminates the potential for violation of the accident analysis bounds. The completion time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the specification limit, power must be reduced, as required by Specification 3.1.H.6, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds. In accordance with Specification 3.1.A.1.f, four reactor coolant pumps must be in operation when Thermal Power is greater than 10% RTP. Therefore, power may be reduced to less than or equal to 10% power if RCS total flow rate is not in accordance with Specification 3.1.H.2. However, it must be verified that operation is in accordance with Specification 3.1.A.1.e which requires at least two reactor coolant pumps to be in operation for Thermal Power greater than 2% RTP.

Surveillance Requirements
A note to Table 4.1-1 requires verification that pressurizer pressure, RCS average temperature, and RCS total flow rate are within the limits of this technical specification (3.1.H). This is required to be performed once per shift.

The frequency for the surveillance for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. A 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify that operation is within safety analysis assumptions.

The frequency for the surveillance for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. A 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify that operation is within safety analysis assumptions.

The surveillance for RCS total flow rate is performed using the installed flow instrumentation. A 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.

References

1. FSAR Chapter 14, "Safety Analysis"

TABLE 4.1-1 (Sheet 1 of 6)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTS OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Test	Remarks
1. Nuclear Power Range	S	D (1) M (3)*	Q (2)** Q (4)	1) Heat balance calibration 2) Bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial offset 4) Signal to Δ T
2. Nuclear Intermediate Range	S (1)	N.A.	P (2)	1) Once/shift when in service 2) Verification of channel response to simulated inputs
3. Nuclear Source Range	S (1)	N.A.	P (2)	1) Once/shift when in service 2) Verification of channel response to simulated inputs
4. Reactor Coolant Temperature	S ## (2)	24M	Q (1)	1) Overtemperature ΔT, overpower ΔT, and low T_{avg} 2) Normal Instrument check interval is once/shift T_{avg} instrument check interval reduced to every 30 minutes when: - T_{avg}-T_{ref} deviation and low T_{avg} alarms are not reset and, - Control banks are above 0 steps
5. Reactor Coolant Flow	S ##	24M	Q	
6. Pressurizer Water Level	S	18M	Q	
7. Pressurizer Pressure	S ##	18M	Q	High and Low

Table Notation

* By means of the movable incore detector system

** Quarterly when reactor power is below the setpoint and prior to each startup if not done previous month.

*** If either an accumulator level or pressure instrument channel is declared inoperable, the remaining level or pressure channel must be verified operable by interconnecting and equalizing (pressure and/or level wise) a minimum of two accumulators and crosschecking the instrumentation.

\# These requirements are applicable when specification 3.3.F.5 is in effect only.

\#\# The "each shift" frequency also requires verification that the DNB parameters (Reactor Coolant Temperature, Reactor Coolant Flow, and Pressurizer Pressure) are within the limits of Technical Specification 3.1.H.

S - Each Shift
W - Weekly
P - Prior to each startup if not done previous week
M - Monthly
NA - Not Applicable
Q - Quarterly
D - Daily
18M - At least once per 18 months
TM - At least every two months on a staggered test basis (i.e., one train per month)
24M - At least once per 24 months
6M - At least once per 6 months

4.3 REACTOR COOLANT SYSTEM (RCS) TESTING

A. Reactor Coolant System Integrity Testing

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To specify tests for Reactor Coolant System integrity after the system is closed following normal opening, modification or repair.

Specification

1. When the Reactor Coolant System is closed after it has been opened, the system will be leak tested at not less than 2335 psig and in accordance with NDT requirements for temperature.

2. When Reactor Coolant System modifications or repairs have been made which involve new strength welds on components, the new welds will meet the requirements of ASME Section XI.

3. The reactor coolant system leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first 11.00 EFPYs of operations. Figure 4.3-1 will be recalculated periodically. Allowable pressures during cooldown from the leak test temperature shall be in accordance with Figure 3.1-2.

Basis

For normal opening, the integrity of the system, in terms of strength, is unchanged. If the system does not leak at 2335 psig (Operating pressure + 100 psi ± 100 psi is normal system pressure fluctuation), it will be leak tight during normal operation.

For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak tightness during normal operation.

The inservice leak test temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, Appendix G. This Code requires that a safety factor of 1.5 times the stress intensity factor caused by pressure be applied to the calculation.

4.3 REACTOR COOLANT SYSTEM (RCS) TESTING

B. Reactor Coolant System Flow Calculation

Specification

Once every 24 months, prior to exceeding 24 hours of continuous operation with THERMAL POWER ≥ 90% RTP, verify by flow calculation that RCS total flow rate is ≥ 385,400 gpm.

Basis

Measurement of RCS total flow rate by performance of a flow calculation once every 24 months verifies that the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The frequency of 24 months reflects the importance of verifying flow after a refueling outage when the core has been altered or steam generator tubes have been plugged, which may have caused an alteration of flow resistance.

This specification allows for placement of the unit in the best condition for performing the Surveillance Requirement. The specification allows the Surveillance Requirement to be performed within 24 hours after THERMAL POWER ≥ 90% RTP. This is appropriate because a flow calculation performed with the plant ≥ 90% RTP will ensure that instrument inaccuracies are consistent with those assumed in the accident analyses. The Surveillance shall be performed within 24 hours of continuous operation at or above 90% RTP.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 INTRODUCTION

In a letter dated March 29, 1996, the Power Authority of the State of New York (PASNY) requested changes to the Indian Point 3 (IP3) Technical Specifications (TSs) to add requirements associated with departure from nucleate boiling (DNB) limits. The proposed changes would specify a limit for minimum steady state operating pressure and for reactor coolant system (RCS) temperature and flow rate. In a subsequent letter dated July 12, 1996, the originally proposed changes to the minimum RCS flow and the maximum RCS average temperature (T_{avg}) were revised due to the assumption of a 100% helium release from the boron coating of the integral fuel burnable absorber (IFBA) rods. The September 6, 1996, submittal contained clarifying information and did not change the staff's initial proposed finding of no significant hazards considerations.

2.0 EVALUATION

As a result of their commitment reported in Licensee Event Report (LER) 95-014, PASNY has proposed a new TS (3.1.H) which would provide requirements for maintaining RCS pressure within the limits assumed in the plant safety analyses. In accordance with this proposed TS, indicated pressurizer pressure would be required to be greater than or equal to 2205 psig. This limit appropriately allows for measurement uncertainty and instrument error and ensures that pressurizer pressure will be in accordance with the safety analyses initial assumptions. The limit on pressurizer pressure would not be applicable during short term operational transients such as a thermal power ramp increase greater than 5% of rated thermal power (RTP) per minute or a thermal power step increase greater than 10% of RTP. These exceptions are consistent with the improved standard TS (STS) presented in NUREG-1431. They represent short-term perturbations where actions to control pressure variations might be counterproductive. Also, since these exceptions represent transients initiated from power levels less than 100% of RTP, an increased margin exists in DNB ratio (DNBR) to offset the temporary pressure variations.

The absorption of neutrons in the thin boride coating on the fuel pellets in an IFBA rod leads to generation of helium as a byproduct. Some of this helium gas can be retained in the pellet coating interface and some can be released into the free space available within the rod. As a result of more recent

Westinghouse test data which indicated that previous helium release assumptions were too low, a reload safety evaluation was performed for Cycle 9 based on a helium release fraction of 100% from IFBA fuel rods. Specifically, the effect of the larger release assumption on the fuel rod design, the thermal-hydraulic DNB analysis, and the large- and small-break loss-of-coolant accident (LOCA) analyses were evaluated.

An analysis of the increased fuel rod internal pressure for Cycle 9 fuel resulting from the increased helium release assumption confirmed that the diametral gap will not increase due to outward cladding creep during steady state operation and extensive DNB propagation will not occur. Therefore, all of the rod internal pressure criteria for which maximum pressure is limiting are satisfied and the effect of the higher helium release assumption on fuel rod design for Cycle 9 operation is acceptable.

For the thermal-hydraulic parameters, the increase in the IFBA helium release fraction improves the pellet-to-cladding heat transfer and reduces the predicted minimum fuel temperature. A minimum fuel temperature evaluation performed for Cycle 9 with the assumption of 100% IFBA helium release showed that the reduction that occurs in the minimum fuel temperatures remained within acceptance criteria.

This application also proposes to relocate the maximum indicated RCS average temperature (T_{avg}) limit from TS 3.1.A.6 to TS 3.1.H and to change the value of T_{avg} from 578.3°F to 571.5°F. Although the existing T_{avg} value of 578.3°F still supports safety analyses for DNB transients (at a minimum measured flow of 385,400 gpm), the post-LOCA containment integrity analyses were performed using a lower value (571.5°F). Therefore, the proposed TS change to reduce the value of T_{avg} from 578.3°F to 571.5°F will make the DNB TS consistent with the more limiting containment integrity analyses and allow the TS to bound all DNB and non-DNB design basis analyses.

For Cycle 9, DNB propagation limits are satisfied for operation up to a burnup of 14,000 megawatt days per metric ton uranium (MWD/MTU) and the current TS minimum measured RCS flow requirement of 332,240 gpm (thermal design flow of 323,600 gpm). For burnups beyond 14,000 MWD/MTU, available DNBR margin was used to show that no rods would experience DNB. The sources of margin include flow margin due to a minimum measured flow of 385,400 gpm (thermal design flow of 375,400 gpm) following installation of new steam generators. Therefore, the proposed TS changes revise the required minimum flow rate to 385,400 gpm to accommodate the assumption of a 100% IFBA helium release fraction. The RCS flow rate limit of 385,400 gpm allows for a measurement uncertainty of 2.6% associated with the performance of the RCS flow calculation required by TS 4.3.B. Although the flow rate is currently stated in TS Figure 2.1-1, it is not clear that this is an operating limit. The relocation to TS 3.1.H clarifies that this flow rate is an operating limit.

PASNY has stated that IFBA fuel pressures in the minimum range may adversely affect the peak clad temperature (PCT) results for the large-break LOCA. However, the accrued Cycle 9 burnup, which already exceeds 7,000 MWD/MTU, ensures that the IFBA pressures are already beyond the range of potential adverse PCT effects for the remainder of Cycle 9 operation. In response to a staff question concerning the effect of higher gap pressure on calculated fuel cladding swelling and rupture during a large-and small-break LOCA PASNY stated in their September 6, 1996, submittal that the greater internal gas pressure leads to increased swelling and increased gap size. This reduces the PCT, the burst temperature and the magnitude of the metal-water reaction temperature excursion.

TS Table 4.1-1 is being revised to specifically state that the parameters for reactor coolant temperature, reactor coolant flow, and pressurizer pressure are to be verified to be within the limits of TS 3.1.H at an "each shift" frequency. Since this requirement is already being met, specifying it in the TS is acceptable and is consistent with the STS.

A proposed revision to TS 4.3 would include the requirement to perform a RCS flow calculation once every 24 months. This is acceptable since this flow calculation is currently performed for each 24-month IP3 cycle after startup from a refueling outage.

The proposed changes clarify existing limits on the measurable parameters (RCS temperature, pressure, and flow rate) so that the resulting DNB value is consistent with initial condition assumptions used in existing safety analyses. The proposed actions, if RCS pressure, RCS T_{avg}, and/or RCS total flow are not within TS limits, are consistent with (or are more conservative than) currently acceptable requirements. In addition, the analyses affected by the assumption of a 100% helium release from the boron coating of the IFBA rods continue to meet the acceptance criteria. Based on the review, we conclude that the proposed revisions are acceptable.

In addition to the DNB limits and RCS flow changes discussed above, the proposed changes modified the Definitions section of the TS by replacing Rated Power with Rated Thermal Power and by adding a definition for Thermal Power. Furthermore, the proposed changes added a note to Figure 2.1-1 to clarify that the pressures and temperatures indicated in that figure do not include allowance for instrument error. These changes are administrative, and appropriately reflect the changes to TS limits and values. Therefore, the staff finds them acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a consideration, and there has been no public comment on such finding (61 FR 37301 and 61 FR 42283). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: L. Kopp

Date: October 22, 1996