ATTACHMENT I TO IPN-96-071

PROPOSED TECHNICAL SPECIFICATIONS FOR A 100% HELIUM RELEASE FROM THE BORON COATING OF THE INTEGRAL FUEL BURNABLE ABSORBER RODS AND A REDUCTION OF MAXIMUM PERMISSIBLE REACTOR COOLANT SYSTEM AVERAGE TEMPERATURE

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

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

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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%, [2]

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.

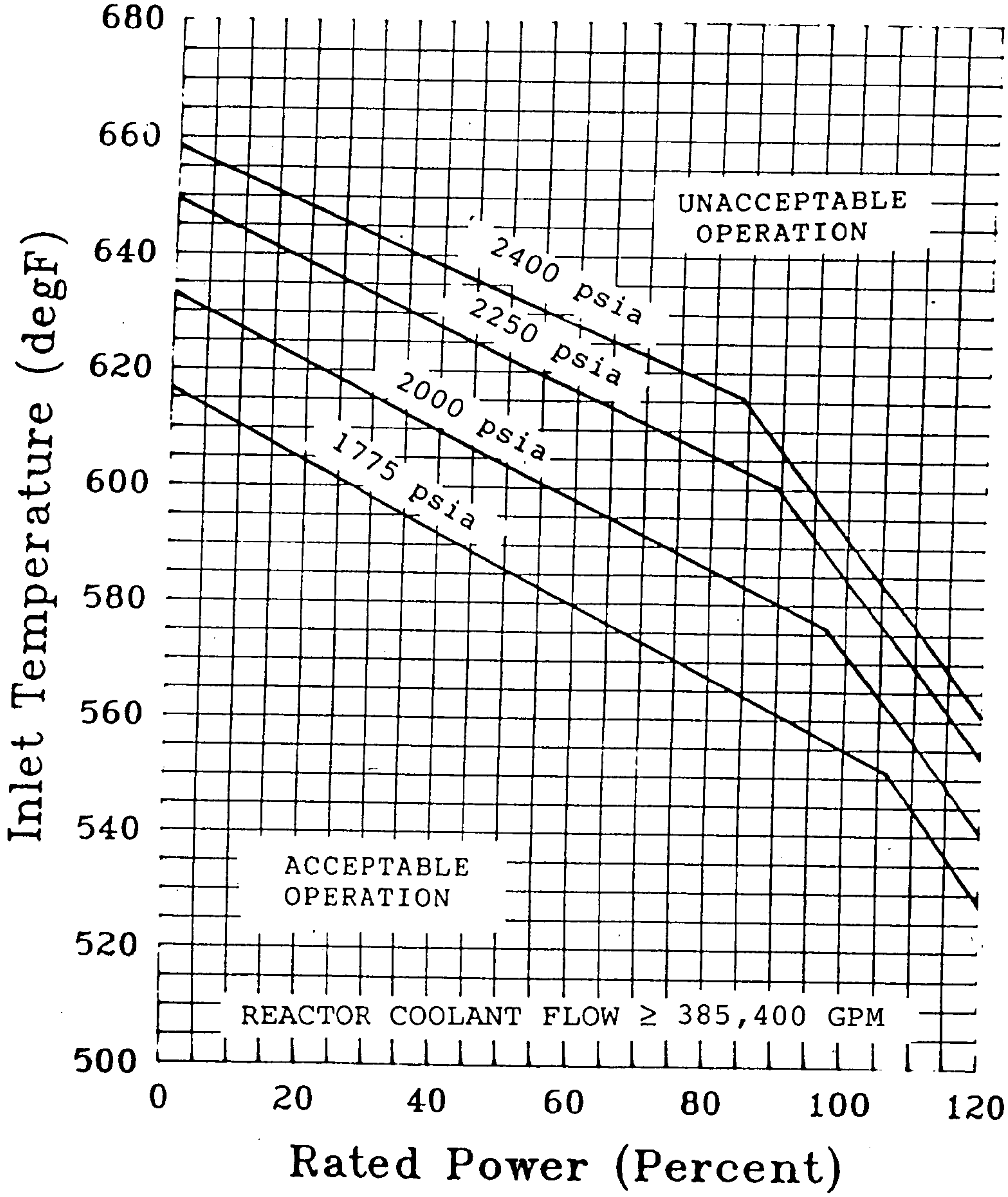
$F_{\Delta H}^{RTP}$ is the $F_{\Delta H}^{N}$ limit at Rated Thermal Power specified in the COLR, and $PF_{\Delta H}$ is the Power Factor Multiplier specified in the COLR.

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

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 $\pm$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_{cold} 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_{cold} of 547.9°F is not exceeded at a measured flow of $\geq$ 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 brought 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 $\pm$ 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.

ATTACHMENT II TO IPN-96-071

SAFETY EVALUATION OF THE PROPOSED TECHNICAL SPECIFICATIONS FOR A 100% HELIUM RELEASE FROM THE BORON COATING OF THE INTEGRAL FUEL BURNABLE ABSORBER RODS AND A REDUCTION OF MAXIMUM PERMISSIBLE REACTOR COOLANT SYSTEM AVERAGE TEMPERATURE

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

SAFETY EVALUATION RELATED TO
PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH A 100% HELIUM RELEASE FROM THE BORON COATING OF THE INTEGRAL FUEL BURNABLE ABSORBER RODS AND A REDUCTION OF MAXIMUM PERMISSIBLE REACTOR COOLANT SYSTEM TEMPERATURE

Section I - Description of Changes

This application describes technical specification changes associated with an assumption of a 100% helium release from the boron coating of the Integral Fuel Burnable Absorber (IFBA) rods. Specifically, the technical specification parameters affected by this application are the minimum reactor coolant system (RCS) flow and the maximum RCS average temperature (T_{avg}).

In order to facilitate review of this submittal, these changes are superimposed onto those submitted by Reference 1 which added requirements associated with Departure from Nucleate Boiling (DNB) limits. Therefore, the technical specification changes contained in Attachment 1 supersede those submitted by Reference 1. The safety evaluation for the changes associated with the DNB limits, submitted as part of Reference 1, is still valid and is not repeated in this application. The safety evaluation and no significant hazards determination presented below specifically address the changes to T_{avg} and to the minimum RCS flow requirement. At the end of Cycle 9, the RCS minimum flow requirement proposed by this application will be changed for Cycle 10, and the changes associated with T_{avg} and those originally proposed by Reference 1 will remain in the technical specifications.

Section II - Evaluation of Changes

The Indian Point 3 Cycle 9 reload safety evaluation was based on a helium release fraction from IFBA fuel rods which has been revised as a result of additional Westinghouse test data which indicated that previous helium release assumptions were too low. Reference 2 analyzed the effects of an assumption of a 100% helium release from the boron coating of the IFBA rods and determined that it potentially affected the fuel rod design, thermal/hydraulic DNB analysis, and large and small break LOCAs. A summary of these results is provided below.

Fuel Rod Design

The increased helium release from the IFBA coating results in increased fuel rod internal pressure predictions. The fuel rod design criteria for which maximum pressure is limiting are the fuel rod internal pressure related criteria which are stated as follows:

> The internal pressure of the lead rod in the reactor will be limited to a value below that which could cause (1) the diametral gap to increase due to outward cladding creep during steady state operation, and (2) extensive DNB propagation to occur.

Rod internal pressure analysis for Cycle 9 fuel, performed with the assumption of a 100% best estimate IFBA helium release, confirm that these rod internal pressure criteria are satisfied.

Thermal/Hydraulic Design

For thermal/hydraulic parameters, the effects on the minimum fuel temperatures and compliance with DNB propagation limits are affected by the use of a 100% IFBA helium release fraction. An evaluation demonstrated that the reduction that occurs in the minimum fuel temperatures remained within acceptance criteria. In addition, DNB propagation limits are satisfied for Cycle 9 up to 14,000 MWD/MTU and a minimum measured flow of 332,240 gpm (current Tech Spec RCS minimum flow requirement). For Cycle 9 beyond 14,000 MWD/MTU, the minimum RCS flow margin must be revised to 385,400 gpm. The proposed Technical Specification changes contained in Attachment I revise this flow rate to accommodate the assumption of a 100% IFBA helium release fraction.

Small and Large Break LOCAs

Reference 2 states that the acceptance criteria of the small and large break LOCAs continue to be met for the assumption of a 100% helium release from the IFBA rods.

RCS Average Temperature

This application proposes to change the value of T_{avg} from 578.3°F to 571.5°F. This change is necessary as the post-LOCA containment integrity safety analyses are performed using a lower T_{avg} (i.e., 571.5°F) than the current Technical Specification value of 578.3°F. The existing value of 578.3°F supports safety analyses for DNB transients and is still valid for these analyses. By reducing the limiting T_{avg} from 578.3°F to 571.5°F, the DNB technical specification will bound all DNB and non-DNB design basis analyses. In the past, the plant has been operated at a lower value than both the existing and proposed T_{avg} values due to procedural requirements and high temperature alarm setpoints. Therefore, the safety analyses bound past operation. This change to T_{avg} will ensure that future plant operation is also within the limits of the safety analyses.

Section III - No Significant Hazards Evaluation

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of any accident previously evaluated?

The proposed changes to the RCS minimum flow and maximum T_{avg} requirements will not increase the probability or consequences of an accident previously evaluated. Reference 2 states that, for the remainder of Cycle 9, all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the Technical Specification Bases is not reduced in any of the licensing basis accident analyses for the assumption of a 100% helium release from the IFBA rods. Reference 3 states that a reduction of maximum allowable indicated T_{avg} from 578.3°F to 571.5°F makes the DNB technical specifications consistent with the more limiting containment integrity analyses. The associated plant and technical specification changes do not affect any of the mechanisms postulated in the FSAR to cause licensing basis events. Therefore, the probability of an accident previously evaluated has not increased. Because design limitations continue to be met, and the integrity of the RCS pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid. Therefore, the consequences of an accident previously evaluated will not be increased.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any previously evaluated?

The proposed changes to the RCS minimum flow and maximum T_{avg} requirements do not create the possibility of a new or different kind of accident from any previously evaluated. Reference 2 states that, for the remainder of Cycle 9, all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the Technical Specification Bases is not reduced in any of the licensing basis accident analyses for the assumption of a 100% helium release from the IFBA. Reference 3 provides clarifications of the assumptions made in the design basis and restricts DNB temperature limits to be consistent with non-DNB analyses. The associated plant and technical specification changes do not change the plant configuration in a way which introduces a new potential hazard to the plant (i.e., no new failure mode has been created). Therefore, an accident which is different than any previously evaluated will not be created.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

The proposed changes to the RCS minimum flow and maximum T_{avg} requirements do not involve a significant reduction in a margin of safety. Reference 2 demonstrates that, for the remainder of Cycle 9, all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the Technical Specification Bases is not reduced in any of the licensing basis accident analyses for the assumption of a 100% helium release from the IFBA. Reference 3 maintains the margin of safety by restricting a DNB limit to bound other analyses. Since References 2 and 3 demonstrate that all applicable acceptance criteria continue to be met, the subject operating conditions will not involve a significant reduction in a margin of safety.

Section IV - Impact of Changes

These changes will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR and SER Conclusions
Overall Plant Operations and the Environment

Section V - Conclusions

The incorporation of these changes: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not significantly reduce the margin of safety as defined in the bases for any technical specification; d) does not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section VI - References

1. NYPA letter (IPN-96-040), W. J. Cahill, Jr. to NRC, "Proposed Changes to Technical Specifications Regarding Departure from Nucleate Boiling Limits," dated March 29, 1996.

2. SECL-96-046, "IFBA Helium Release Evaluation For Cycle 9 Restart," Westinghouse Electric Corporation, dated July 8, 1996.

3. Westinghouse letter, "Technical Specification Value for T-Average," INT-96-557, dated July 3, 1996.

ATTACHMENT III TO IPN-96-071

AUTHORITY COMMITMENTS FOR THE PROPOSED TECHNICAL SPECIFICATIONS FOR A 100% HELIUM RELEASE FROM THE BORON COATING OF THE INTEGRAL FUEL BURNABLE ABSORBER RODS AND A REDUCTION OF MAXIMUM PERMISSIBLE REACTOR COOLANT SYSTEM AVERAGE TEMPERATURE

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

COMMITMENTS ASSOCIATED WITH IPN-96-071

Comm. No.	Commitment Description	Due Date
IPN-96-071-01	Revise procedures to relect the new RCS flow and T_{avg} requirement.	10/25/96
IPN-96-071-02	Revise FSAR.	Next applicable annual update