123 Main Street
White Plains, New York 10601
914-681-6840
914-287-3309 (FAX)

NewYorkPower Authority

William J. Cahill, Jr.
Chief Nuclear Officer

September 6, 1996
IPN-96-098

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Response to Request for Additional Information Concerning Proposed Technical Specification Changes 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

References:

1. NRC letter, G. F. Wunder to W. J. Cahill, Jr., regarding a request for additional information on the helium release/T_{avg} proposed technical specification, dated July 31, 1996.

2. NYPA letter, W. J. Cahill, Jr. to NRC, dated July 11, 1996, "Proposed Changes to Technical Specifications Regarding 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," (IPN-96-071).

Dear Sir:

This letter provides the Authority's response to the NRC's request for additional information (Reference 1). The request concerns the Authority's proposed Technical Specification changes regarding a 100% helium release from the boron coating of the Integral Fuel Burnable Absorber rods and a reduction of the maximum permissible reactor coolant system average temperature (Reference 2). The NRC's questions followed by the Authority's responses are contained in Attachment I to this letter.

In addition, Attachment II contains a Westinghouse report (SECL-96-046, "IFBA Helium Release Evaluation For Cycle 9 Restart," dated July 8, 1996). This document was listed as an enclosure to Reference 2, but was inadvertently omitted from the package submitted on July 11, 1996.

This letter contains no new commitments. If you have any questions, please contact Ms. C. D. Faison.

Very truly yours,

William J. Cahill, Jr.
Chief Nuclear Officer

Attachments: as stated

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

Resident Inspector's Office
Indian Point Unit 3
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, NY 10511

Mr. George F. Wunder, Project Manager
Project Directorate I-1
Division of Reactor Projects I/II
U.S. Nuclear Regulatory Commission
Mail Stop 14B2
Washington, DC 20555

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

ATTACHMENT I TO IPN-96-098

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING PROPOSED CHANGES TO 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

Response to Request for Additional Information

This attachment provides the Authority's response to the NRC's request for additional information. The request concerns the Authority's proposed Technical Specification changes 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. The NRC's questions are followed by the Authority's responses.

NRC Question 1

What is the effect of higher gap pressure on the calculated fuel cladding swelling and rupture during large and small break loss of coolant accidents (LOCAs)?

NYPA Response

The limiting burnup condition for the integral fuel burnable absorber (IFBA) rods analyzed in the Indian Point 3 large break LOCA (LBLOCA) is beginning of life (BOL). The BOL condition minimizes the rod internal pressure of the IFBA fuel rod, which yields a higher peak cladding temperature (PCT). For the LBLOCA transient, lower rod internal pressure in the IFBA fuel rod results in a higher cladding temperature at the time of fuel rod burst, which increases the PCT due to the more severe metal-water reaction. Therefore, the higher rod internal pressure resulting from the increased helium release, combined with the increase in pressure from the accrued burnup on the fuel, would decrease the IFBA PCT. A general discussion of the effect of higher rod internal pressure on the LBLOCA transient follows.

The increase in gap pressure would result in more swelling of the fuel rod during the LBLOCA transient, which would increase the gap size, reducing the cladding temperature. This effect, however, is overshadowed by the effect of fuel rod burst and blockage, which is discussed below.

Higher gap pressure in the IFBA fuel rod would result in a decrease in the fuel rod burst temperature in the LBLOCA analysis, which in turn results in a lower IFBA PCT. The rupture of a fuel rod is primarily a function of the differential pressure across the cladding (system pressure minus rod internal pressure), and the temperature of the cladding. An increase in the rod internal pressure would increase the differential pressure across the cladding during the refill and reflood periods. With a higher differential pressure, a lower cladding temperature would be required to burst the rod. Since the metal-water reaction induced temperature excursion that accompanies fuel rod burst is highly dependent upon the temperature of the cladding at the time of burst, the PCT calculated for the IFBA fuel rod would decrease for Indian Point 3 as a result of the lower burst temperature. This is the dominant phenomena for the LBLOCA transient.

Another phenomenon which can affect the LBLOCA PCT is that of flow blockage in the hot channel resulting from fuel rod swelling and burst. Fuel rod swelling and burst can result in a reduction in the flow area in the channel, and subsequent reduction in heat transfer. The amount

of blockage calculated to occur is also a function of the burst temperature. Although the flow area reduction is not a linear function of burst temperature, for typical LBLOCA transients a high burst temperature will result in a larger reduction in the flow area. Therefore, an increase in the rod internal pressure would reduce the burst temperature, which would tend to reduce the blockage in the range of IFBA pressures. For analyses in which the PCT occurs later in the transient, the PCT would be reduced. Although blockage is considered in the analysis, the LBLOCA transient is dominated by the metal-water reaction temperature increase associated with fuel rod rupture.

The higher gap pressure can affect the small break LOCA (SBLOCA) analysis in two ways, both of which do not have significant effects on the calculated PCT. For SBLOCA transients in which fuel rod burst is of no concern (i.e., those in which the cladding temperature remains below the threshold temperature for significant metal-water reaction), an increase in gap pressure will tend to reduce the PCT by a small amount. This reduction is due to the larger gap caused by increased swelling.

The Westinghouse methodology for SBLOCA transients in which fuel rod burst is of concern involves a calculation of the PCT increase resulting from fuel rod rupture at the appropriate burnup condition. The PCT increase is calculated at a burnup that results in a rod internal pressure just high enough to burst the fuel rod, given the cladding temperature transient. In SBLOCA analyses, the burst temperature is not significantly affected by changes in the burnup condition and/or rod internal pressure. Thus, the temperature increase resulting from the metal-water reaction is also not significantly affected.

NRC Question 2

If the calculated swelling or rupture increases, what are the effects on the calculated oxidation and PCT (10 CFR 50.46 (b))?

NYPA Response

The response to question 1 also answers this question.

Summary

In conclusion, all of the parameters cited by the NRC have been appropriately addressed in the analysis, and none of them result in conditions that would place the plant outside the bounds of its analytical design basis. In summary:

> The increased helium release results in incrementally larger internal fuel rod gas pressures throughout the life of the fuel rod. 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. This is primarily a LBLOCA effect and has little influence on SBLOCA results. Furthermore, the design basis LBLOCA is at BOL (when internal rod pressure is at a minimum), which is more limiting than at end of life (when internal pressure is greatest due to the 100% helium release.)

ATTACHMENT II TO IPN-96-098

SECL-96-046
IFBA HELIUM RELEASE EVALUATION FOR CYCLE 9 RESTART

New York Power Authority
INDIAN POINT 3 NUCLEAR POWER PLANT
Docket No. 50-286

Customer Reference No(s). N/A

WESTINGHOUSE NUCLEAR SAFETY
SAFETY EVALUATION CHECK LIST (SECL)

1) NUCLEAR PLANT(S): Indian Point - Unit 3

2) SUBJECT (TITLE): IFBA Helium Release Evaluation For Cycle 9 Restart

3) The written safety evaluation of the revised procedure, design change or modification required by 10 CFR 50.59 (b) has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A 10 CFR 50.59(a)(1)

(3.1) Yes__ No X A change to the plant as described in the FSAR?
(3.2) Yes__ No X A change to procedures as described in the FSAR?
(3.3) Yes__ No X A test or experiment not described in the FSAR?
(3.4) Yes X No__ A change to the plant Technical Specifications? (See Note on Page 2.)

4) CHECK LIST - PART B 10 CFR 50.59(a)(2) (Justification for Part B answers must be included on page 2.)

(4.1) Yes__ No X Will the probability of an accident previously evaluated in the FSAR be increased?
(4.2) Yes__ No X Will the consequences of an accident previously evaluated in the FSAR be increased?
(4.3) Yes__ No X May the possibility of an accident which is different than any already evaluated in the FSAR be created?
(4.4) Yes__ No X Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
(4.5) Yes__ No X Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
(4.6) Yes__ No X May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
(4.7) Yes__ No X Will the margin of safety as defined in the Bases to any Technical Specification be reduced?

NOTES:

If the answer to any of the above questions is unknown, indicate under Section 5.0 REMARKS and explain below.

If the answer to any of the above questions in Part A (3.4) or Part B cannot be answered in the negative, based on written safety evaluation, the change review would require an application for license amendment as required by 10 CFR 50.59(c) and submitted to the NRC pursuant to 10 CFR 50.90.

5) REMARKS:

The answers given in Section 3, Part A, and Section 4, Part B, of the Safety Evaluation Checklist, are based on the attached Safety Evaluation.

FOR FSAR UPDATE

Technical Specifications mark-ups attached:

Please note that the attached Technical Specification package is the same package attached to SECL-96-072, Safety Evaluation Of Cycle 9 Low Pressure Operation (July 1996). This is being done in order to provide a consistent set of mark-ups for ease of review because some of the same mark-ups apply to both safety evaluations. The mark-ups that specifically apply to this safety evaluation are so noted. They are located as follows: Section 2.1 (page 2.1-2), Figure 2.1-1, Section 3.1 H (pages 3.1-36,-37), and Section 4.3.B. (page 4.3-4).

---

6) SAFETY EVALUATION APPROVAL LADDER:

Prepared By: L.V.Tomasic Date: 7-3-96

Reviewed By: R.R.Laubham Date: 7-8-96

1.0 SUMMARY

This safety evaluation addresses an assumption of a 100% Helium Release from the boron coating of the Integral Fuel Burnable Absorber (IFBA) rods for the restart of Cycle 9, and for completing Cycle 9 operation, supports the attached Technical Specification changes, and, concludes that there is no unreviewed safety question pursuant to 10CFR 50.59, (a), (2), criteria.

For Cycle 9 restart, for burnup from the accrued current burnup of 7,000 MWD/MTU up to 14,000 MWD/MTU, and a minimum measured flow of 332,240 (thermal design flow of 323,600 gpm), current DNB propagation limits are satisfied, and the Cycle 9 Reload Safety Evaluation remains valid. For burnups beyond 14,000 MWD/MTU, DNBR margin was used to show that no rods would be in 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 the new steam generators. The minimum measured flow uncertainty at 100% power is 2.6%, which has been verbally confirmed by NYPA as being consistent with their 24 month fuel cycle surveillance requirements. For Cycle 9 operation beyond 14,000 MWD/MTU, a minimum measured flow of 385,400 gpm is required and the attached Technical Specifications have been marked-up to reflect this flow for Cycle 9 operation completion. (Since Technical Specification, Section 2.3, Specification 1.B.6 (a), for Power Reactor Coolant Loop Low Flow Setpoint being equal to, or greater than, 90% of normal indicated loop flow does not state a flow quantity, no mark-up is provided).

2.0 INTRODUCTION

The Indian Point Unit 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 test data. Notification of the change and the preliminary impacts were presented to the New York Power Authority (NYPA) in Westinghouse letter 96 IN-G-005, dated February 22, 1996. The following addresses those criteria impacted by the change in the helium release fraction. The evaluation is valid from the current burnup, approximately 7000 MWD/MTU, until the end of Cycle 9.

3.0 LICENSING BASIS

Title 10 of the Code of Federal Regulations, Part 50.59 (10 CFR 50.59) allows the holder of a license authorizing operation of a nuclear power facility the capacity to initiate certain changes, tests, and experiments not described in the Final Safety Analysis Report (FSAR). Prior Nuclear Regulatory Commission (NRC) approval is not necessary for implementation provided that the change, test, or experiment does not involve an unreviewed safety question or change in the Technical Specifications incorporated in the license. It is, however, the obligation of the licensee to maintain records of changes, tests, and experiments to the facility to the extent that

such changes impact the FSAR. 10CFR50.59 further stipulates that these records shall include a written safety evaluation which provides the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question.

4.0 EVALUATION

The following safety related areas and analyses may be affected by the assumption of a 100% Helium Release from the boron coating of the IFBA rods, and are addressed in this safety evaluation.

Fuel Rod Design
Thermal / Hydraulic DNB Analysis
Large Break (LB) LOCA
Small Break (SB) LOCA
Technical Specifications

The following safety related areas and analyses are not affected by the assumption of a 100% Helium Release from the boron coating of the IFBA rods, and are not addressed in this safety evaluation.

Nuclear Design
Non-LOCA Analyses
Main Steamline Break (MSLB) Mass and Energy Release
LOCA Hydraulic Forces
Post-LOCA Long Term Cooling
Hot Leg Switch Over
Containment Integrity Analyses
Steam Generator Tube Rupture
Safety Systems Setpoints
Mechanical Components and Systems
Instrumentation and Control Systems
Emergency Operating Procedures

4.1 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.

The effects of assuming 100% best estimate IFBA helium release fraction on rod internal pressure criteria margins have been evaluated for Indian Point Unit 3 Cycle 9 fuel operation. Rod internal pressure analysis for Indian Point Unit 3 Cycle 9 fuel, performed with the assumption of 100% best estimate IFBA helium release, confirm that part (1) of the rod internal pressure criterion is satisfied.

Part (2) of this criterion, which addresses DNB propagation, has also been satisfied for Indian Point Unit 3 Cycle 9 operation. For Cycle 9 burnup up to 14,000 MWD/MTU, current Westinghouse DNB propagation limits are satisfied. For cycle burnup beyond 14,000 MWD/MTU, it has been shown that there is sufficient excess DNB margin to the 95/95% DNB design basis to confirm that an initial DNB event will not occur during a Condition II event. Since DNB propagation depends on the occurrence of a rod initially in DNB, demonstrating that DNB will not occur assures that DNB propagation will not occur. For Condition III/IV events, the only event for which DNB is expected to occur is the ejected rod event, and it has been shown that the maximum fraction of the core in DNB, including the effects of DNB propagation, is less than the current limit of 15% assumed for this event.

For some fuel rod design criteria, such as transient clad stress and transient clad strain, clad fatigue, and maximum fuel temperature, increased IFBA helium release and the associated increase in rod internal pressure results in increased design margin, and therefore these criteria are not adversely affected by the change in the IFBA helium release fraction and have not been specifically reevaluated. Other fuel rod design criteria are relatively insensitive to rod internal pressure, such as steady state clad strain, clad corrosion, clad flattening, plenum spring support, fuel rod growth and clad free standing. These criteria are unaffected by the change in the IFBA helium release fraction and have also not been reevaluated.

In summary, all fuel rod design limits are satisfied for Indian Point Unit 3 Cycle 9 operation with 100% IFBA helium release.

4.2 Thermal/Hydraulic Design

For the thermal-hydraulic parameters, the minimum fuel temperatures are affected by the use of 100% IFBA helium release fraction; the maximum fuel temperatures are not affected. The increase in the IFBA helium release improves the pellet-to-cladding heat transfer and reduces the predicted minimum fuel temperatures (minus uncertainties). A minimum fuel temperature evaluation was performed for Indian Point 3 Cycle 9 with the assumption of 100% IFBA helium release. The resulting reductions in the minimum fuel average temperature and in the fuel minimum surface temperature remain within the acceptance limits.

For Cycle 9 burnup up to 14,000 MWD/MTU and a minimum measured flow of 332,240 gpm, current DNB propagation limits are satisfied, and the Cycle 9 Reload Safety Evaluation remains valid. For burnups beyond 14,000 MWD/MTU, DNBR margin was used to show that no rods would be in DNB. Margin sources include flow margin due to a minimum measured flow of 385,400 gpm following installation of the new steam generators.

4.3. Large Break LOCA

For the large break LOCA, IFBA fuel pressures in the minimum range may adversely affect the peak clad temperature (PCT) results. However, the accrued burnup (7000 MWD/MTU) at the time of shutdown ensures that the IFBA pressures are already beyond the range of potential adverse effects for the remainder of Cycle 9 operation.

4.4 Small Break LOCA

Although IFBA pressure uncertainties are modeled in the small break LOCA analysis, the small break LOCA analysis has been generically evaluated as not being adversely affected by the changes in IFBA pressure uncertainties. Therefore, the acceptance criteria of the small break LOCA continue to be met.

4.5 Technical Specifications

For Cycle 9 burnups beyond 14,000 MWD/MTU, this evaluation requires increasing the minimum measured flow to 385,400 gpm (see attached mark-ups).

5.0 DETERMINATION OF UNREVIEWED SAFETY QUESTION

The evaluation of the assumption of a 100% Helium release from the IFBA rods concludes that it will not result in a potential unreviewed safety question, as defined in 10CFR50.59, (a),(2).

5.1 Will the probability of an accident previously evaluated in the FSAR be increased?

The assumption of a 100% Helium release from the IFBA rods does not result in a condition where the material, and construction standards, which were applicable prior to the change are altered. System integrity is maintained. The modification does not cause the initiation of any accident nor create any new credible limiting single failure nor result in any event previously deemed incredible being made credible. The existing separation of the control and protection functions are not adversely impacted. In addition, the safety

functions of safety related systems and components, which are related to accident mitigation, have not been altered. Therefore, the probability of an accident previously evaluated in the FSAR will not be increased by the assumption of a 100% Helium release from the IFBA rods.

5.2 Will the consequences of an accident previously evaluated in the FSAR be increased?

The assumption of a 100% Helium release from the IFBA rods does not affect the integrity of the fuel assembly or reactor internals such that its function in the control of radiological consequences is affected. In addition, the assumption of a 100% Helium release from the IFBA rods does not affect any fission barrier. The assumption of a 100% Helium release from the IFBA rods does not change, degrade, or prevent the response of safety related mitigation systems to accident scenarios, as described in the FSAR. In addition, there is no affect on any assumption previously made in the radiological consequence evaluations nor affect on the mitigation of the radiological consequences of an accident described in the FSAR. Therefore, the consequences of an accident previously evaluated in the FSAR will not be increased.

5.3 May the possibility of an accident which is different than any previously evaluated in the FSAR be created?

The assumption of a 100% Helium release from the IFBA rods would not cause the initiation of any accident nor create any new credible limiting single failure. The assumption of a 100% Helium release from the IFBA rods would not result in any event previously deemed incredible being made credible. In addition, the safety functions of safety related systems and components, which are related to accident mitigation, have not been altered. As such, it does not create the possibility of an accident different than any evaluated in the FSAR.

5.4 Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

The assumption of a 100% Helium release from the IFBA rods would not result in an increased probability of scenarios previously deemed improbable. It does not create any new failure modes for the safety-related equipment. The assumption of a 100% Helium release from the IFBA rods would not result in any original design specification, such as seismic requirements, electrical separation requirements and environmental qualification, being altered. In addition, the assumption of a 100% Helium release from the IFBA rods would not result in equipment used in accident mitigation to be exposed to an adverse environment. Therefore, the assumption of a 100% Helium release from

the IFBA rods would not increase the probability of a malfunction of equipment important to safety previously evaluated in the FSAR.

5.5 Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

The assumption of a 100% Helium release from the IFBA rods would not result in a different response of safety-related systems and components to accident scenarios than that postulated in the FSAR. No new equipment malfunctions have been introduced that will affect fission product barrier integrity. In addition, there is no affect on any assumption previously made in the radiological consequence evaluations nor affect on the mitigation of the radiological consequences of an accident described in the FSAR. Therefore, the assumption of a 100% Helium release from the IFBA rods would not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR.

5.6 May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

The assumption of a 100% Helium release from the IFBA rods would not create failure modes that could adversely impact safety-related equipment, nor cause the initiation of any accident. The assumption of a 100% Helium release from the IFBA rods would not result in any event previously deemed incredible being made credible. In addition, the safety functions of safety related systems and components, which are related to accident mitigation, have not been altered. Therefore, it will not create the possibility of a malfunction of equipment important to safety different than previously evaluated in the FSAR.

5.7 Will the margin of safety as defined in the bases to any Technical Specifications be reduced?

The assumption of a 100% Helium release from the IFBA rods will have no affect on the availability, operability, or performance of the safety-related systems and components.

For Cycle 9 restart, for a burnup from the accrued current 7,000 MWD/MTU up to 14,000 MWD/MTU, and a minimum measured flow of 332,240 gpm, current DNB propagation limits are satisfied, and the Cycle 9 Reload Safety Evaluation remains valid. For burnups beyond 14,000 MWD/MTU, DNBR margin was used to show that no rods would be in DNB. Margin sources include flow margin due to a minimum measured flow of 385,400 gpm following installation of the new steam generators.

6.0 CONCLUSION

The analyses affected by the assumption of a 100% Helium Release from the boron coating of the IFBA rods were evaluated. It has been determined that these analyses continue to meet the analyses acceptance criteria. The analyses evaluated include Fuel Rod Design, Thermal / Hydraulic DNB Analysis, Large Break LOCA, and Small Break LOCA.

Therefore, it has been concluded that an assumption of a 100% Helium Release from the boron coating of the Integral Fuel Burnable Absorber (IFBA) rods for the restart of IP-3, Cycle 9, would support the attached Technical Specification changes, and, does not constitute an unreviewed safety question pursuant to 10CFR 50.59, (a), (2), criteria.

ATTACHMENT to SECL-96-046, REV 0

PAGE 1 OF 19

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.

Changes proposed by NYPA to NRC (=PN-96-040)

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.

correlations which have

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.

Design

Amendment No. 48, 61, 86,

Replace with Insert A

~~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:

SECL 94-046

1. $F_{\Delta H}^{RTP} = F_{\Delta H}^{N}$ limit at Rated Thermal Power (RTP) specified in the COLR.
2. ~~an equivalent~~ the existing steam generator tube plugging level ~~based on tube-to-tube~~ for cycle 9. ~~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 ~~[illegible]~~ 385,400 gpm as ~~measured~~ indicated 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:

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

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 reactor coolant system 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.

Insert A to pg 2.1-2

In meeting the DNB design criterion, uncertainties in operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined such that there is at least a 95% probability at a 95% confidence level that the minimum DNBR of the limiting fuel rod is greater than or equal to the DNBR limit of the DNB correlation being used.

Additional DNBR margin is maintained by performing the safety analysis to a higher DNBR limit. This margin between the Design and Safety Analysis Limit DNBR value is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

References

1. FSAR Section 3.2.2

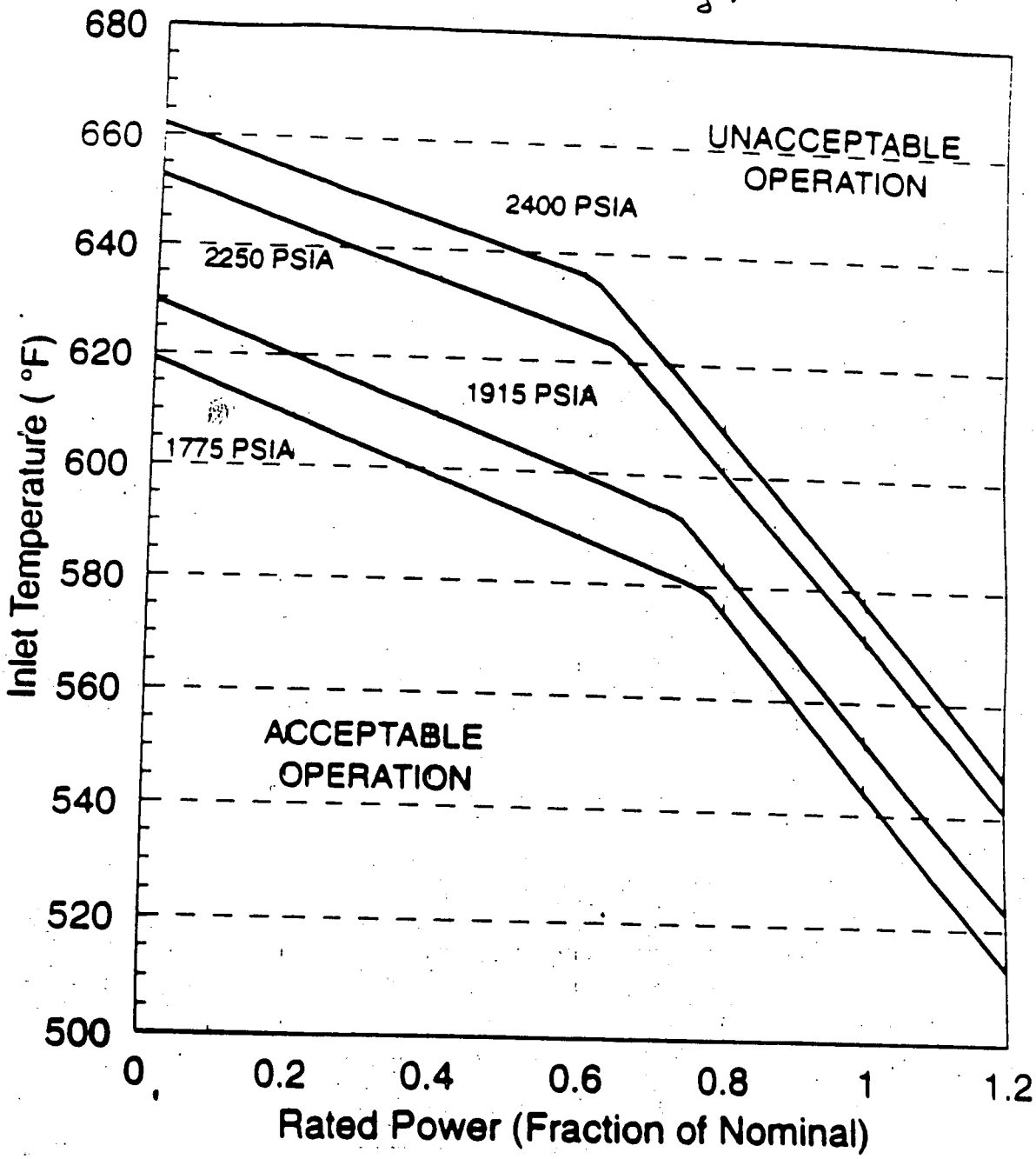
2. "Safety Evaluation for Indian Point Unit 3 with Asymmetric Tube Plugging Among Steam Generators", WCAP-10705 (Westinghouse Non-Proprietary), October 1984.

3. "IFBA Helium Release Evaluation for Cycle 9 Restart", SECL-96-046, May 1996.

4. "Safety Evaluation of Cycle 9 Low Pressure Operation SECL-96-072, May 1996.

REACTOR CORE SAFETY LIMITS

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



100 PERCENT RATED POWER IS EQUIVALENT TO 3025 MWT

Based on a Reactor Coolant Flow of greater than or equal to 385400 gpm

Pressures and temperatures do not include allowance for instrument error.

Figure 2.1-1

SECL 96-046

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, and pressurizer level.

Objective

To provide for automatic protective action such that the principal process variables do not exceed a safety limit.

Specification

1. Protective instrumentation for reactor trip settings shall be as follows:

   A. Startup protection

      (1) High flux, power range (low setpoint) - ≤25% of rated power

   B. Core limit protection

      (1) High flux, power range (high setpoint) - ≤109% of rated power.

      (2) High pressurizer pressure - ≤2385 psig.

      (3) Low pressurizer pressure - ≥1800 psig.

      (4) Overtemperature ΔT

      $$\Delta T \leq \Delta T_o \left[K_1 - K_2 (T_{avg} - T') + K_3 (P - P') - f(\Delta I)\right]$$

Add an additional specification to identify a high flux, power range (high setpoint) trip value applicable to operation at 85% RTP and 1900 psig during Cycle 9 only. This trip value must be consistent with a Safety Analysis Limit of 100.3% of the current 3025 MWt RTP.

Reviewers Note

NYPA must define the trip setpoint based on this data

ΔT_o ≤ Measured full power ΔT for the channel being calibrated, °F

T_{avg} = Average Temperature for the channel being calibrated, °F (input from instrument racks)

T' = Measured full power T_{avg} for the channel being calibrated, °F

P = Pressurizer pressure, psig (input from instrument racks)

P' = 2235 psig (i.e., nominal pressurizer pressure at rated power)

See Reviewer's Note on Previous Page

K_1 ≤ 1.35

Add an additional specification to identify the K_1 value applicable to operation at 85% RTP and 1900 psig during Cycle 9 only. This added spec must be consistent with a Safety Analysis Limit of 1.265.

K_2 = 0.0212

K_3 = 0.000981

K_1 is a constant which defines the overtemperature ΔT trip margin during steady state operation if the temperature, pressure and f(ΔI) terms are zero.

K_2 is a constant which defines the dependence of the overtemperature ΔT setpoint to T_{avg}

K_3 is a constant which defines the dependence of the overtemperature ΔT set point to pressurizer pressure.

ΔI = $q_t - q_b$, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power.

f(ΔI) = 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, where q_t and q_b are defined above such that:

(a) for $q_t - q_b$ below 6 percent, f(ΔI) = 0.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.6 percent of rated power.

Add an additional specification to identify the K_4 value applicable to operation at 85% RTP and 1900 psig during Cycle 9 only. This added spec must be consistent with a Safety Analysis Limit of 1.00.

See Reviewer's Note on 2.3-1

(5) Overpower ΔT

$$\Delta T \leq \Delta T_o \left(K_1 - K_2 \frac{dT_{avg}}{dt} - K_3(T_{avg} - T')\right)$$

where:

ΔT_o ≤ measured full power ΔT for the channel being calibrate °F

T_{avg} = measured average temperature for the channel bei calibrated, °F (input from instrument racks)

T' = measured full power T_{avg} for the channel being calibrate °F (can be set no higher than 573.3 °F)

K_1 ≤ 1.073

K_2 = 0 for decreasing average temperature

≥ 0.175 sec/°F for increasing average temperature

K_3 = 0 for $T \leq T'$

≥ 0.00116 for $T > T'$

K_1 is a constant which defines the overpower ΔT trip sar during steady state operation if the temperature term zero.

K_2 is a constant determined by dynamic considerations compensate for piping delays from the core to the l temperature detectors; it represents the combination the equipment static gain setting and the time const setting.

K_3 is a constant which defines the dependence of overpower ΔT setpoint to T_{avg}.

$\frac{dT_{avg}}{dt}$ = rate of change of T_{avg}

(6) Low reactor coolant loop flow:

(a) ≥ 90% of normal indicated loop flow
(b) Low reactor coolant pump frequency - ≥ 57.2 cps

(7) Undervoltage - ≥ 70% of normal voltage

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.

Reviewer's Note: Markup applies prior to NRC approval of IPN-96-010

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.

Deletion proposed by NYPA to NRC (IPN-96-010)

6. **Reactor Coolant System T_{avg}**

During steady state operation (at 100% RTP and 2235 psig), the maximum indicated T_{avg} shall not exceed 578.3°F. During steady ~~state~~ operation at 85% RTP & 1900 psig (Cycle 9 only), the maximum indicated T_{AVG} shall not exceed Y °F.

Amendment No. 73, 81, 88, 121

See Insert B for proposal

I. Reactor Coolant System RCS

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

Specification

1. During the POWER OPERATION CONDITION, with four reactor coolant pumps running, RCS DNB parameters for pressurizer pressure, ~~and~~ RCS average temperature, and RCS ~~total~~ flow rate shall be within the limits specified below:

Replace with table identified as Insert B. SECL 96-046

a. Pressurizer pressure ≥ 2235 psig;

b. Maximum indicated T_{avg} ≤ 578.3°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 ≥ 332,240 gpm.

2 ~~b~~. 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.

3 ~~c~~. 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.2 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.

4 ~~d~~. 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.

5 ~~e~~. If RCS total flow rate is not restored to within the limits of Specification 3.1.H.1 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 (References 1, 2, and 3) 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

INSERT B

ITEM	2235 PSIG NOMINAL OPERATING PRESSURE	1900 PSIG NOMINAL OPERATING PRESSURE
a. Nominal Core Power	≤ 100% RTP	≤ 85% RTP
b. Minimum Pressuizer Pressure	≥ 2205 psig	≥ X psig ~~(based on a Safety Analysis Limit of 1900 psig)~~
c. Maximum Indicated T_{AVG}	≤ 578.3°F	≤ Y°F ~~Based on a Safety Analysis Limit of 570.5°F and uniform SGTP)~~
d. Minimum RCS flow	≥ 385,400 gpm	≥ 385,400 gpm

SECL 96-044

X, Y : *NYPA must establish values for these parameters based on applicable plant uncertainties*

and nominal pressure of 1900 psig and nominal T_{AVG} of 570.5°F and uniform SGTP.

3.1.H continued:

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 sounds 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)~~ (References 1, 2, and 3). 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

Specification 3.1.H.1 ~~and 3.1.H.2~~ specifies 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 ~~[illegible]~~ 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 ~~[illegible]~~ 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.

SECL 96-046

SECL 96-046

The pressurizer pressure limits of ~~[illegible]~~ allow for measurement uncertainty and instrument error. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit.

Add ~~another value~~ 2205 psig (based on nominal operating conditions of 100% RTP and 2235 psig) and X psig (based on nominal operating conditions of 85% RTP and 1900 psig during Cycle 9 only)

The limits on maximum indicated RCS average temperature provide 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

3.1.H continued

Plugging among steam generators. A maximum full power T[illegible] of 54[illegible] F including control deadband and measurement uncertainties was assumed in these safety analyses. Restricting the maximum indicated T[illegible] to 578.1 F assures that a T[illegible] of 547.9 F is not exceeded at a measured flow of 331,240 gpm when considering asymmetric tube plugging among steam generators.

Insert C

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.2 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.3 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.

Insert C to pg 3.1-38

For cycle 9 only, the limit on RCS flow is based on existing steam generator tube plugging levels and does not include the historical allowance for future steam generator tube plugging. The cycle 9 limit is intended to maximize DNB margin during the cycle to maximize the power limit for reduced pressure operation.

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.X5 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 11% RTP. Therefore, power may be reduced to less than or equal to 11% power if RCS total flow rate is not in accordance with Specification 3.1.H.X1 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 11% 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"

2. "IFBA Helium Release Evaluation for Cycle 9 Restart", SECL-96-046, May 1996.

3. "Safety Evaluation of Cycle 9 Low Pressure Operation", SECL-96-072, May 1996.

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 ≥ ~~332,240~~ 385,400 gpm. SECL 96-046

Basis

Measurement of RCS total flow rate by performance of a flow [illegible] every 24 months verifies [illegible] 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.

Note: This Surveillance added by NYPA to NRC IPN-96-040

Amendment No.