Docket No. 50-286

August 27, 1985

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Mr. John C. Brons
Senior Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

The Commission has issued the enclosed Amendment No.61 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated March 27, 1985 and April 23, 1985, as supplemented August 1, 1985.

The amendment revises the Technical Specifications to allow the first of a three-phase fuel design transition from Westinghouse 15 X 15 low parasitic (LOPAR) design to the 15 X 15 Optimized Fuel Assembly (OFA) design with the introduction of Wet Annular Burnable Absorber (WABA) rods into the core and to allow 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%.

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

Sincerely,

/s/JDNeighbors

Joseph D. Neighbors, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 61 to DPR-64
2. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:DL	ORB#1:DL	ORB#1:DL	BC-ORB#1:DL	OELD	AD-OR:DL
CParrish	for KJohnston;ps	JDNeighbors	for SVarga	MKarman	GLainas
8/15/85	8/15/85	8/15/85	8/15/85	8/21/85	8/21/85

Subject to changes to be made

Changes made

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PDR ADOCK 05000286
P PDR

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of New York

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

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. 61
License No. DPR-64

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

   A. The applications for amendment by Power Authority of the State of New York (the licensee) dated March 27, 1985 and April 23, 1985, as supplemented August 1, 1985, comply 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:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 61 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 27, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 61

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.1-1	2.1-1
2.1-2	2.1-2
2.1-3	———
Figure 2.1-1	Figure 2.1-1
2.3-2	2.3-2
———	2.3-2a
2.3-3	2.3-3
2.3-5	2.3-5
2.3-6	2.3-6
2.3-7	2.3-7
2.3-8	———
3.1-2	3.1-2
3.1-3	3.1-3
———	3.1-3a
3.10-1	3.10-1
3.10-7	3.10-7
3.10-8	3.10-8
3.10-9	3.10-9
3.10-11	3.10-11
3.10-13	3.10-13
3.10-15	3.10-15
Figure 3.10-2	Figure 3.10-2
5.3-1	5.3-1

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.

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 W-3 L-Grid DNB correlation for Westinghouse LOPAR fuel and the WRB-1 correlation for 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 percent probability with 95 percent confidence that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (1.30 for W-3 L-Grid correlation and 1.17 for WRB-1 correlation). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit[1].

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 design 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. $F^N_{\Delta H}$ of 1.55

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 323,600 gpm[2], and

4. a reference cosine with a peak of 1.55 for axial power shape[3].

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

$$F^N_{\Delta H} \leq 1.55\,[1 + 0.3\cdot(1-P)]$$

Where P is the fraction of Rated Thermal Power.

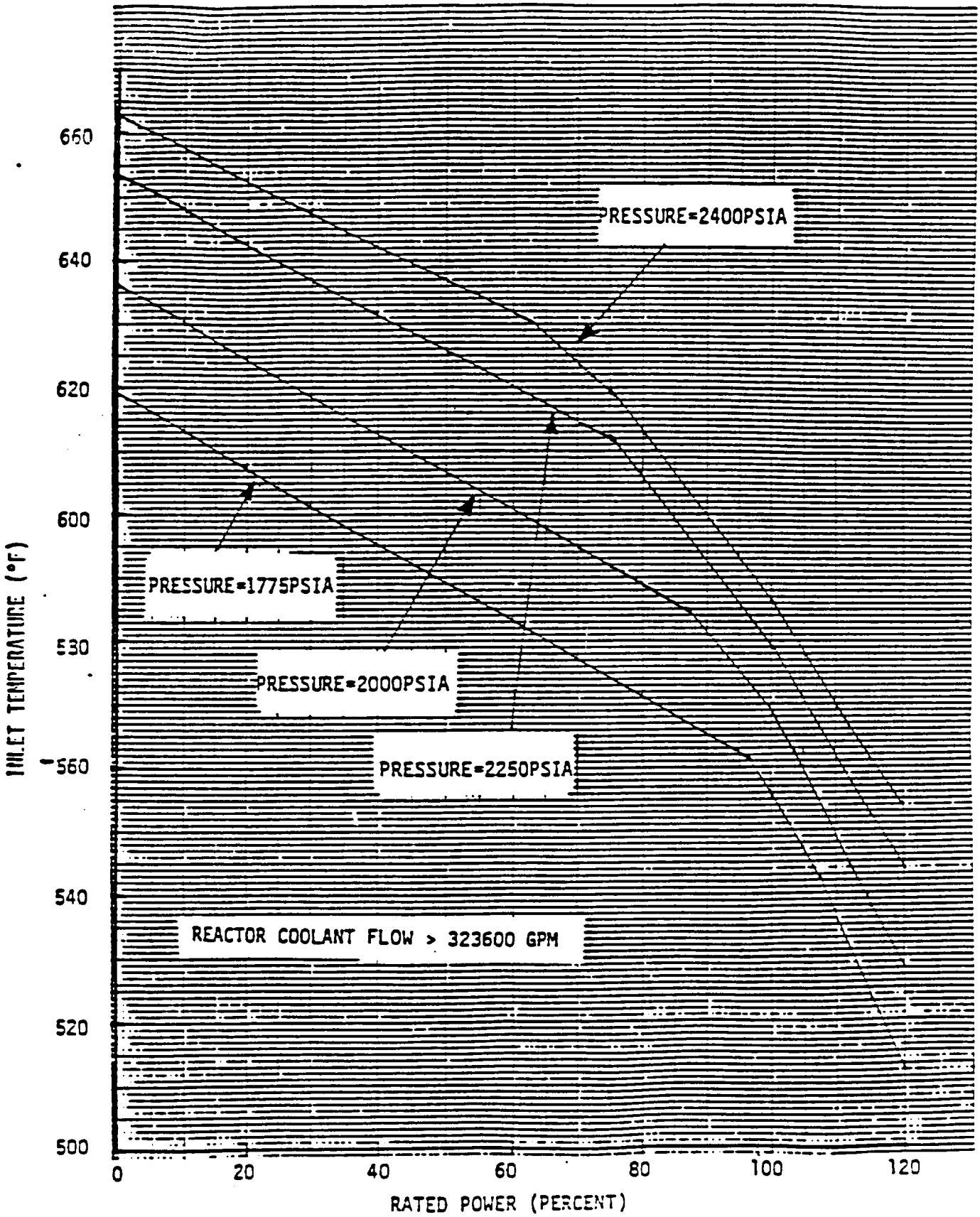
These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit (Figure 3.10-4) assuming the axial power imbalance is within the limits of the $f(\Delta I)$ function of the Overtemperature ΔT trip. 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.

<u>References</u>

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. Fuel Densification - Indian Point Nuclear Generating Unit No. 3, WCAP-8147 (Westinghouse Non-Proprietary), July 1973.



100 PERCENT RATED POWER IS EQUIVALENT TO 3025 MWT

FIGURE 2.1-1

Amendment No. ~~48~~, 61

ΔT_o = Indicated ΔT for the channel being calibrated at rated power, °F

T_{avg} = Average Temperature for the channel being calibrated, °F

T' = Indicated T_{avg} for the channel being calibrated at nominal rated power conditions, °F

P = Pressurizer pressure, psig

P' = Indicated nominal pressurizer pressure at rated power = 2235 psig

K_1 $\leq$ 1.135

K_2 = 0.0114 $\pm$ .00057

K_3 = 0.00066 $\pm$ .0000033

K_1 is a constant which defines the over temperature 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 set point 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.

$f(\Delta I)$ = 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 $q_t - q_b$ within -28, + 5 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +5 percent, the ΔT trip set point shall be automatically reduced by an equivalent of 2.3 percent of rated power.

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -28 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.1 percent of rated power.

(5) Overpower ΔT

$$\Delta T \leq \Delta T_o \left(K_4 - K_5 \frac{dT_{avg}}{dt} - K_6 (T_{avg} - T') - f(\Delta I)\right)$$

where

ΔT_o	=indicated ΔT for the channel being calibrated at rated power, F
T_{avg}	=measured average temperature for the channel being calibrated, °F
T'	= indicated T_{avg} for the channel being calibrated at nominal rated conditions, °F
K_4	≤ 1.089
K_5	= 0 for decreasing average temperature ≥ 0.175 sec/°F for increasing average temperature
K_6	= 0 for $T \leq T'$ ≥ 0.00116 for $T > T'$
K_4	is a constant which defines the overpower ΔT trip margin during steady state operation if the temperature and the f (ΔI) terms are zero.
K_5	is a constant determined by dynamic considerations to compensate for piping delays from the core to the loop temperature detectors; it represents the combination of the equipment static gain setting and the time constant setting.
K_6	is a constant which defines the dependence of the overpower ΔT setpoint to T_{avg}
$f(\Delta I)$	= as defined above.
$\frac{dTavg}{dt}$	= rate of change of T_{avg}

(6) Low reactor coolant loop flow:
(a) $\geq$ 90% of normal indicated loop flow
(b) Low reactor coolant pump frequency - $\geq$55.0 cps

(7) Undervoltage - $\geq$70% of normal voltage

The source and intermediate range reactor trips do not appear in the specification as these settings are not used in the transient and accident analysis (FSAR Section 14). Both trips provide protection during reactor startup. The former is set at about 10^{+5} counts/sec and the latter at a current proportional to approximately 25% of rated full power.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is backed up by the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss of coolant accident. Its setting limit is consistent with the value assumed in the loss of coolant analysis.[4]

The overtemperature Delta-T reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 3.5 seconds)[5], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors[2], is always below the core safety limit as shown on Figure 2.1-1. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.[6][7] The values of the constants K_1, K_2, and K_3 are determined during the design of the core for operation with all reactor loops in service. The values are then specified for the reactor protection system manufacturer and for calibration. The setpoints will ensure that the safety limit of centerline fuel melt will not be reached and the applicable design limit DNBR will not be violated.

The overpower Delta-T reactor trip prevents power density anywhere in the core from exceeding 112% of design power density, as described in Section 7.2.3 and 14.1.2 and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation (via the overall gain in the rate controller) for piping delays from the core to the loop temperature detectors. The specified set points meet this requirement and include allowance for instrument errors.[2] The values of the constants K_4, K_5, and K_6 are determined during the design of the core and the reactor protection system. The values are then specified for the reactor protection system manufacturer and for calibration.

The overpower and overtemperature protection system setpoints include the effects of fuel densification on core safety limits. The revised setpoints as given above will ensure that the combination of power, temperature, and pressure will not exceed the revised core safety limits as shown in Figures 2.1-1.

The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. Fuel temperature decreases due to cladding creepdown with burnup and consequential reduction of pellet-cladding gap. Thus overpower limits become less restrictive as fuel burnup proceeds.

The constants delta-T_0 and T' for each overtemperature and overpower protection channel are set in accordance with the indicated delta-T and Tavg at rated power existing in the loop from which the process inputs for a particular protection channel are supplied. This is done to account for loop to loop differences in delta-T and Tavg which may exist as a result of asymmetric steam generator tube plugging.

The low flow reactor trip protects the core against DNB in the event of a loss of one or two reactor coolant pumps. The undervoltage reactor trip protects the core against DNB in the event of a loss of two or more reactor coolant pumps. The set points specified are consistent with the values used in the accident analysis.(8) The low frequency reactor coolant pump trip also protects against a decrease in flow. The specified set point assures a reactor trip signal by opening the reactor coolant pump breaker before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1600 ft^3 of water (39.75 ft. above the lower instrument tap) corresponds to 92% of span. The specified set point allows margin for instrument error and transient level overshoot beyond their trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against postulated loss of feedwater accidents. The specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the Auxiliary Feedwater System.(9)

Specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set points at which these trips are unblocked assures their availability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above the P-8 setpoint for four-loop operation, an automatic reactor trip will occur if any pump is lost. This latter trip will prevent the minimum value of the DNB ratio, DNBR, from going below the applicable design limit during normal operational transients.

The turbine and steam-feedwater flow mismatch trips do not appear in the specification as these settings are not used in the transient and accident analysis (FSAR Section 14).

References

(1) FSAR 14.1.1

(2) FSAR 14.1.2

(3) FSAR Table 14.1-1

(4) FSAR 14.3.1

(5) FSAR 14.1.2

(6) FSAR 7.2

(7) FSAR 3.2.1

(8) FSAR 14.1.6

(9) FSAR 14.1.9

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 valves 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. Reactor Coolant System Tavg

During steady state operation, the maximum indicated Tavg shall not exceed 576°F.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only (1), hence, the specified upper limit of 2% rated power without operating pumps provides a substantial safety factor.

The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensee and approved for less than four loop operation at power levels above 10% rated power has been granted by the Commission. (See license condition 2.C.(3)).

Each of the pressurizer code safety valves is designed to relieve 420,000 lbs. per hr. of saturated steam at the valve set point.

If no residual heat were removed by the Residual Heat Removal System, the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for overpressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load (2) without a direct reactor trip or any other control.

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.

The limit on maximum indicated Tavg provides assurance that Reactor Coolant System Temperatures are maintained within the normal steady-state envelope of operation assumed in the FSAR transient and accident analyses and in WCAP-10704, "Safety Evaluation of Indian Point Unit 3 with Asymmetric Tube Plugging Among Steam Generators." WCAP-10704 assumed a maximum full-power T_{cold} of 546.9°F (including control deadband and measurement uncertainties). As shown in Tables II-1 and II-2 of WCAP-10704, a maximum indicated Tavg of 576°F (including 2°F measurement uncertainty) is calculated for a full power

T_{cold} of 546.9°F at a flow of 323,600 gpm. Restricting maximum Tavg to 576°F (indicated) at all power levels will preserve the steady-state DNB margins assured in WCAP-10704.

References

1) FSAR Section 14.1.6

2) FSAR Section 14.1.8

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the limits on core fission power distribution and to limits on control rod operations.

Objectives:

To ensure:

1. Core subcriticality after reactor trip,
2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and
3. Limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications:

3.10.1 Shutdown Reactivity

The shutdown margin shall be at least as great as shown in Figure 3.10-1.

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$F_Q(Z) \leq (2.13/p) \times K(Z)$ for $P > 0.5$

$F_Q(Z) \leq (4.26) \times K(Z)$ for $P \leq 0.5$

$F^N_{\Delta H} \leq 1.55\ [1 + 0.3\ (1-P)]$

Where P is the fraction of full power at which the core is operating, K(Z) is the fraction given in Figure 3.10-2 and Z is the core height location of F_Q.

3.10.6.3 If a control rod having a rod position indicator channel out of service, is found to be misaligned from 3.10.6.1a above, then Specification 3.10.5 will be applied.

3.10.7 Inoperable Rod Limitations

3.10.7.1 An inoperable rod is a rod which does not trip or which is declared inoperable under Specification 3.10.5 or fails to meet the requirements of 3.10.8.

3.10.7.2 Not more than one inoperable control rod shall be allowed any time the reactor is critical except during physics tests requiring intentional rod misalignment. Otherwise, the plant shall be brought to the hot shutdown condition.

3.10.7.3 If any rod has been declared inoperable, then the potential ejected rod worth, associated transient power distribution peaking factors and the accident listed in Table 3.10-1 shall be analysed within 5 days, or the reactor brought to the hot shutdown condition using normal operating procedures. The analysis shall include due allowance for non-uniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

3.10.8 Rod Drop Time

At operating temperature and full flow, the drop time to each control rod shall be no greater than 2.4 seconds from loss of stationary gripper coil voltage to dashpot entry.

3.10.9 Rod Position Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift and after a load change greater than 10 percent of rated power.

3.10.10 Reactivity Balance

The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm 1\%$ $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall, at least consider reactor coolant system boron concentration, control rod position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration. The predicated reactivity values shall be adjusted (normalized) to correspond to the actual core condition prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

3.10.11 Notification

Any event requiring plant shutdown on trip setpoint reduction because of Specification 3.10 shall be reported to the Nuclear Regulatory Commission within 30 days.

Basis

Design criteria have been chosen for normal operations, operational transients and those events analyzed in FSAR Section 14.1 which are consistent with the fuel integrity analysis. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also, the minimum DNBR in the core must not be less than the applicable design limit DNBR in normal operation or in short term transients.

In addition to the above conditions, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant

F^E_Q Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

An upper bound envelope of 2.13 times the normalized peaking factor axial dependence of Figure 3.10-2 has been determined consistent with Appendix K criteria and is satisfied for OFA transition mixed cores [3] by all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on this upper bound normalized envelope of Figure 3.10-2 demonstrates a peak clad temperature not greater than 1995°F, which is below peak clad temperature limit of 2200°F.[2]

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_{\Delta H}$ there is a 8 percent allowance for uncertainies which means that normal operation of the core is expected to result in $F^N_{\Delta H} < 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape

4. Axial Power Distribution Control Procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In Specification 3.10.2, F_Q is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The referenced value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that F_Q upper bound envelope of 2.13 times Figure 3.10-2 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the control rod bank more than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup

-11 percent indicated) increasing by ±1 percent for each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90 percent or lower, depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the ±5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essense of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the boron system to position the control rods to produce the required indicated flux difference.

For FSAR Section 14.1 events, the core is protected from overpower and a minimum DNBR of the applicable design limit DNBR by an automatic protection system. Compliance with operating procedures is assumed as a precondition for FSAR Section 14.1 events. However, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Quadrant power tilt limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation, as this phenomenon is caused by some asymmetric perturbation, e.g., rod misalignment, or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication System or core instrumentation per Specification 3.10.6, and core limits are protected per Specification 3.10.5. A quadrant tilt by some other means would not appear instantaneously, but would build up over several hours and the quadrant tilt limits are met to protect against this situation. They also serve as a backup protection against the dropped or misaligned rod. Operational experience shows that normal power tilts are less than 1.01. Thus, sufficient time is available to recognize the presence of a tilt and correct the cause before a severe tilt could build up. During startup and power escalation, however, a large tilt could be initiated. Therefore, the Technical Specification has been written so as to prevent escalation above 50 percent power if a large tilt is present. The numerical limits are set to be commensurate with design and safety limits for DNB protection and linear heat generation rate as

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequence of a hypothetical rod ejection accident. The available control rod reactivity, or excess beyond needs, decreases with decreasing boron concentration because the negative reactivity required to reduce the core power level from full power to zero is largest when the boron concentration is low.

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.4) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod, that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determiniations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worth. These measurements will augment the normal fuel cycle design calaculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

The rod position indicator channel is sufficiently accurate to detect a rod ±7 inches away from its demand position. An indicated misalignment less than 12 steps does not exceed the power peaking factor limits. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or moveable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 12 step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothethical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 5 day period is short compared with the time interval required to achieve a significant, non-uniform fuel depletion.

The required drop time to dashpot entry is consistenmt with safety analysis.

<u>REFERENCE</u>

1. WCAP-8576, "Augmented Startup and Cycle 1 Physics Program:, August 1975
2. FSAR Appendix 14C
3. Letter from J. P. Bayne to S. A. Varga dated April 23, 1985, entitled "Proposed Technical Specifications Regarding the Cycle 4/5 Refueling".

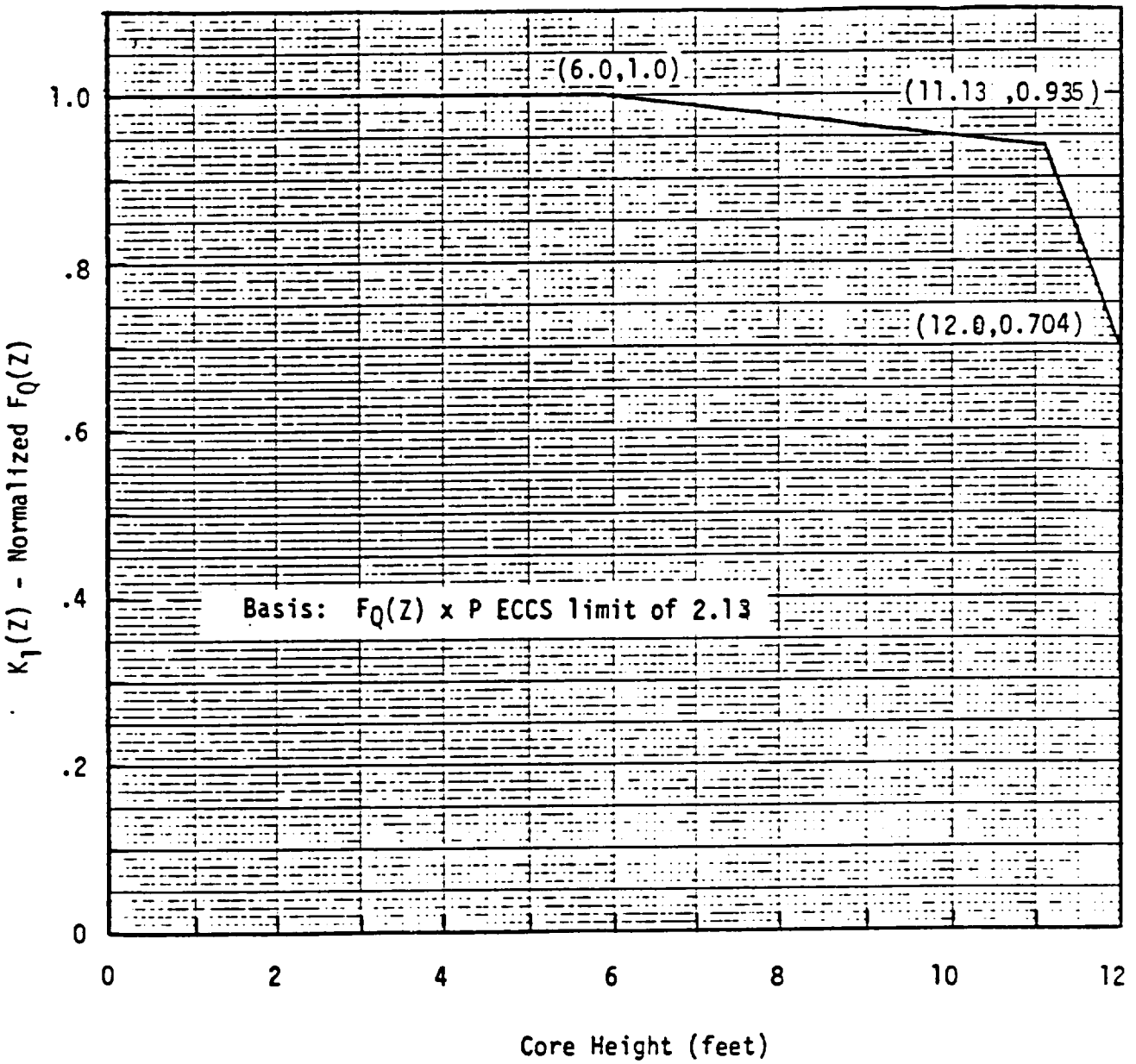


FIGURE 3.10-2 Hot Channel Factor Normalized Operating Evenlope

5.3 REACTOR

Applicability

Applies to the reactor core, and reactor coolant system.

Objective

To define those design features which are essential in providing for safe system operations.

A. Reactor Core

1. The reactor core contains approximately 87 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods.(1)

2. The average enrichment of the initial core was a nominal 2.8 weight percent of U-235. Three fuel enrichments were used in the initial core. The highest enrichment was a nominal 3.3 weight percent of U-235.(2)

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 3.4 weight percent of U-235.

4. Burnable poison rods were incorporated in the initial core. There were 1434 poison rods in the form of 8, 9, 12, 16, and 20-rod clusters, which are located in vacant rod cluster control guide tubes.(3) The burnable poison rods consist of borosilicate glass clad with stainless steel.(4) Burnable poison rods of an approved design may be used in reload cores for reactivity and/or power distribution control.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 61 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

By application dated April 23, 1985, New York Power Authority (the licensee) requested (Ref. 1) an amendment to the Technical Specifications contained in Appendix A of Facility Operating License No. DPR-64. The Technical Specification changes are intended to accommodate: (1) a planned fuel design change from the Westinghouse 15 X 15 low parasitic (LOPAR) design to the 15 X 15 Optimized Fuel Assembly (OFA) design, (2) planned use of Wet Annular Burnable Absorber (WABA) rods, and (3) use of a modified $F_{\Delta H}$ limit equation. By application dated March 27, 1985, the licensee requested to change the Technical Specifications to allow 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 is less than or equal to 24%. The request was supported by Westinghouse analyses, WCAP-10704 "Safety Evaluation for Indian Point Unit 3 with Asymmetric Tube Plugging Among Steam Generators (Proprietary)" and WCAP -10705 "Safety Evaluation for Indian Point Unit 3 with Asymmetric Tube Plugging Among Steam Generators (Non-Proprietary)" which are reviewed in this evaluation. An April 23, 1985 submittal included a safety evaluation (as Attachment II to Reference 1) that addressed mechanical, nuclear, thermal-hydraulic, and accident analysis considerations. The NRC had previously approved operation with a maximum 24% tube plugging in any steam generator (Ref. 3). An August 1, 1985 submittal provided supplemental information (Ref. 7) only and did not change the proposed Technical Specifications.

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2.0 REFUELING

2.1 FUEL MECHANICAL DESIGN

Indian Point Unit 3 has been operating with all Westinghouse 15 x 15 LOPAR fuel. The Unit 3 Cycle 5 core will receive Westinghouse 15 x 15 OFAs resulting in approximately a 40% OFA/60% LOPAR mixed core. Subsequent reloads are expected to eventually contain only OFA fuel. Although the Westinghouse 15 x 15 OFA is a new design, it is very similar to the 15 x 15 LOPAR fuel design. The major change introduced by the 15 x 15 OFA design is the use of 5 intermediate Zircaloy grids replacing 5 intermediate Inconel grids in the LOPAR fuel. The Zircaloy grids have thicker and wider straps than the Inconel grids in order to closely match the Inconel grid strength. Furthermore, the 15 x 15 OFA Zircaloy grid design is similar to the 17 x 17 OFA grid design, which was described in WCAP-9500-A. This report has been reviewed and approved by the NRC staff (Ref. 4).

A new feature to be introduced with the OFA fuel is the Westinghouse Annular Burnable Absorber (WABA). Also, new thimble plugs will be used to accommodate a 13 mil ID and OD reduction in the guide thimble above the dashpots. The reduction in OFA thimble diameter is due to the use of the Zircaloy grid straps that are thicker than the LOPAR Inconel straps.

Because of the thicker and wider OFA Zircaloy grid straps, the OFA reportedly has about a 4.5% increase in hydraulic resistance to flow compared to a LOPAR assembly. This results in an increased lift force for the OFA and requires the use of 3-leaf holddown springs in the top nozzle instead of the 2-leaf springs used for LOPAR assemblies.

The OFA reconstitutable bottom nozzle feature is similar to that introduced in other Westinghouse plants such as Trojan, Farley Units 1 and 2, Salem Unit 1, North Anna Units 1 and 2, Zion Units 1 and 2, etc. In this design a locking cup is used, instead of a lockwire used in LOPAR assemblies, to lock the thimble screw of a guide thimble tube in place.

The mechanical design requirements and criteria which have been approved for the 17 x 17 OFA design are met by the 15 x 15 OFA design except for a change in the fuel assembly nonoperational 6g loading design basis (Section 4.2.1.5.1.a

of WCAP 9500). The 6g value has been changed to 4g axial and 6g lateral. We find this acceptable.

The OFAs are designed to be compatible with the LOPAR assemblies, reactor internals interfaces, and fuel handling equipment. The grid elevations for the two assembly designs are identical, thus minimizing mechanical and hydraulic interaction. Impact tests have indicated that the Zircaloy grid strength data at reactor operating conditions is structurally acceptable. The reduction in OFA thimble diameter still provides adequate nominal diametral clearance for control rods and other core components. The control rod drop time, however, has been increased from 1.8 seconds for the LOPAR assembly to 2.4 seconds for the OFA as a result of the reduced guide tube clearance. This has been accounted for in the safety analyses. The 3-leaf spring design has been successfully used in the 17 x 17 OFA demonstration program and other 15 x 15 LOPAR assemblies and is fully compatible with the LOPAR assembly and the handling tools at the Indian Point plant.

Based on the above and on the previous approval of Westinghouse 15 x 15 OFA and WABA rods for use in Zion Units 1 and 2, D.C. Cook Unit 1, and Turkey Point Units 3 and 4, we find the use of the 15 X 15 OFA and WABA rods acceptable for Indian Point Unit 3.

2.2 NUCLEAR DESIGN

The nuclear evaluation of the transition from LOPAR to OFA fueled cores has been performed with the Westinghouse Reload Safety Methodology which has been used in previous Indian Point reload analyses and approved by the NRC (Ref. 5). The results show that the expected values of the nuclear parameters fall within the normal cycle-to-cycle variations.

As mentioned in Section 5.0 of this SER, the large break loss of coolant analysis requires a reduction in F_Q from 2.14 to 2.13 during the OFA-LOPAR transition cores. The licensee, in accordance with the requirements of WCAP-9272 (Ref. 5), has confirmed that Westinghouse has evaluated the approved subset of the 18 Case Final Acceptance Criteria (FAC) power distribution analysis indicating that the plant can be safely operated within the newly specified power distribution limits.

On the basis that the nuclear evaluation has been performed with previously accepted methods we conclude that it is acceptable.

2.3 THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic analysis of the mixed OFA-LOPAR core was performed using the same methods described in the FSAR for the 15 x 15 LOPAR fuel except that the WRB-1 critical heat flux (CHF) correlation was used for the OFA and the W-3 L-grid CHF correlation was used for the LOPAR fuel. The W-3 correlation has a design limit DNBR of 1.30 whereas the WRB-1 correlation has a DNBR limit of 1.17. Westinghouse has performed a statistical analysis which provided the basis for the applicability of the WRB-1 correlation for the 15 x 15 OFA. This was found to be acceptable by the NRC for Zion Units 1 and 2, Turkey Point Units 3 and 4, and D.C. Cook Unit 1 and we, therefore, find the WRB-1 correlation acceptable for the 15 x 15 OFA in Indian Point Unit 3.

The thermal-hydraulic analysis of a transitional mixed OFA-LOPAR core has been previously reviewed by the staff and approved with a condition requiring a penalty on DNBR to account for the uncertainty associated with the interbundle cross-flow in the mixed core. The licensee has performed an analysis to determine the required penalty factor in the same manner approved for the 17 x 17 OFA-LOPAR mixed core analysis. The result showed that a 3% penalty is required for the transitional mixed core. This 3% penalty is the same as that approved by the NRC for the Zion and Turkey Point OFA-LOPAR mixed cores and will be required until an eventual full core of OFA fuel is achieved.

The WABA poison rod design is described in WCAP-10021, Revision 1 (Ref. 6) which has been approved by the staff. In order to ensure no violation of the total core bypass flow limit, the total number of WABA rods in the core should be less than the upper limit established in Table 7.2 of WCAP-10021, Revision 1. The licensee has indicated that a total of 752 WABA rods will be used in the Indian Point Unit 3 Cycle 5 core. This number is below the allowed

limit and is, therefore, acceptable. For future reload cores, the number of WABA rods will also be required to be within the allowed limit.

For the LOPAR fuel, the design minimum DNBR limit of 1.30 using the W-3 L-grid CHF correlation is 4.8% higher than the allowable DNBR limit of 1.24 derived from the 15 x 15 L-grid CHF test data. The analysis contains an inherent DNBR margin of 18.1%. This margin is composed of 4.8% from conservatism in the 1.30 DNBR limit, 3.3% from pitch reduction, 3.0% from conservatism in the thermal diffusion coefficient, and 7.0% from a densification model that eliminates the power spike effect on DNB. This DNBR margin is more than sufficient to compensate for the maximum rod bow penalty of 14.9%. For the 15 x 15 OFA fuel, a plant-specific safety analysis DNBR limit of 1.43 was used. This safety analysis DNBR limit has an 18.1% DNBR margin compared with the DNBR limit of 1.17 for the WRB-1 CHF correlation. This 18.1% margin is sufficient to account for the rod bow penalty of 14.9% as well as the transitional mixed core penalty of 3%.

2.4 ACCIDENT AND TRANSIENT EVALUATION

The accidents analyzed in the Indian Point Unit 3 FSAR which could potentially be affected by the OFA reload have been reviewed by the licensee. The increase in rod drop time from 1.8 to 2.4 seconds, as discussed in Section 2.0 of this SER, could affect the fast transients for which the protection system trips the reactor within a few seconds. The evaluation showed that all accidents and transients except the loss of flow, locked rotor, and rod ejection are insignificantly affected by the increased rod drop time. Reanalysis of these three events also showed that all the safety limits and criteria are still satisfied for the increased rod drop time associated with the OFA core. The licensee has stated that these accidents are bounding with respect to either assumption of 24% uniform steam generator tube plugging or 30% asymmetric tube plugging.

The nominal conditions considered for the transition design include a core power of 3025 MWt, system pressure of 2250 psia, core inlet temperature of 542.9°F, and RCS thermal design flow of 323,600 gpm. The worst steady-state operation (102% x 3025 MWt, 323,600 gpm, 2220 psia, and 546.9°F) were used as the initial conditions for the accident analyses. The reactor vessel flow of 323,600 gpm equals the thermal design flow established for the approved 24% uniform tube plugging analysis. Although loop flows will be asymmetric with the assumption of asymmetric tube plugging as high as 30%, the sum of all loop flows were assumed to equal 323,600 gpm (Ref. 2). The tube plugging distribution analyzed was 30%, 24%, 24% and 8%. The flow for each loop was calculated based on the reactor coolant pump head curve and the loop flow impedance, including the effect of plugged tubes in the steam generator.

The blowdown and reflood portions of a postulated large break loss of coolant accident (LOCA) are impacted by the difference in fuel assembly flow resistance between the LOPAR and OFA designs. For the transition mixed core, the licensee has determined the effect of the flow redistribution due to the hydraulic resistance mismatch in the mixed core configuration. Since the 15 x 15 OFA increases the flow resistance by about 4.5%, the reflood flow rate for the 15 x 15 OFA fuel during the transitional mixed core period will be reduced by approximately 2.2%. This results in about a 10°F increase in peak clad temperature (PCT). Since there is insufficient ΔPCT margin in the approved Indian Point Unit 3 LOCA analysis to account for the 10°F increase in PCT, the LOCA F_Q will be reduced by 0.01 (2.14 to 2.13) during the OFA-LOPAR transition cores. However, once a full core of 15 x 15 OFA is achieved, operation at the approved LOCA analysis F_Q of 2.14 will be valid without a mixed core penalty. These restrictions are not impacted by the asymmetry of the tube plugging.

2.5 TECHNICAL SPECIFICATIONS

Changes have been proposed to the Indian Point Unit 3 Technical Specifications in order to allow for the introduction of OFA into the core, the use of WABA rods, a modified $F_{\Delta H}$ limit equation, and up to 30% asymmetric steam generator tube plugging. Each proposed change is discussed below:

Specification 2.1

This specification has been modified to reference RCS vessel "inlet" temperature rather than "average" temperature. With normal (symmetric) plant operation, plant control and protection limits are based on average vessel temperature. With asymmetric tube plugging, however, inlet temperature is a more appropriate limit and must not exceed the value assumed in the analyses. This change is, therefore, acceptable.

Bases for Specification 2.1

The modifications to the Basis include revisions to the $F^N_{\Delta H}$ limit equation, incorporation of the design limit DNBR for OFA, and the limitations of the asymmetric tube plugging evaluation. The increase in the limit equation power multiplier from 0.2 to 0.3 has been previously reviewed and approved by the NRC for several other operating Westinghouse plants such as Trojan, Zion, and McGuire. The use of two different DNBR design limits has been discussed and approved in Section 4.0 of this SER. The allowance of a 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% is reviewed and approved in the appendix to this SE.

Figure 2.1-1

This figure has been revised to incorporate new core limits for the OFA fuel transition as discussed in this SER and is acceptable.

Specifications 2.3.1.B.(4) and 2.3.1.B.(5)

Modifications to these specifications have been proposed in order to revise the definitions to account for different individual loop temperatures as a result of asymmetric tube plugging. In addition, the $f(\Delta I)$ definitions are being revised to reflect the change in the $F^N_{\Delta H}$ multiplier and the new core limits for the OFA fuel transition and are, therefore, acceptable.

Basis 2.3

Reference to three loop operation has been deleted since this operation is not permitted. Modifications have also been made to incorporate the design limit DNBR for OFA. These changes are acceptable.

Specification 3.1.A.6 and Basis 3.1

Specification 3.1.A.6 and the associated Basis has been added to account for a maximum indicated average RCS temperature of 576°F to preserve the steady state DNBR margins. This is conservative with respect to the value (576.7°F) which was assumed in the asymmetric tube plugging analyses (Reference 2). The addition is, therefore, acceptable.

Specification 3.10.2.1

The $F^N_{\Delta H}$ limit equation has been modified as discussed earlier. The slightly reduced 2.13 F_Q limit accounts for a 0.01 F_Q penalty during operation with OFA-LOPAR mixed cores as approved in Sections 3.0 and 5.0 of this SER. These changes are, therefore, acceptable.

Specification 3.10.8

This specification has been modified to account for the increase in rod drop time to 2.4 seconds as a result of the smaller diameter guide tubes in the OFA. Since those accidents which could be significantly affected by the increased rod drop time were reanalyzed to account for the increased drop time and found to meet all safety limits and criteria, the change is acceptable.

Basis 3.10

The basis has been changed to reflect the fact that two different fuel types having different DNBR limits are possible. Also included is the reduced 2.13 F_Q limit, the reference to OFA transition mixed cores, and an acceptable PCT

not greater than 1995°F due to the 0.01 F_Q reduction. These are acceptable changes.

Figure 3.10-2

This figure has been revised to correspond to the reduced 2.13 F_Q limit for OFA transition mixed cores and is acceptable.

Specification 5.3.A

The changes specify the initial core enrichments and type of burnable poison rods. Allowance is also provided for the use of other types of burnable poison rods of an approved design such as WABA rods. This is acceptable.

2.6 SUMMARY

We have reviewed the proposed changes to Technical Specifications involving the use of 15 x 15 OFAs, WABAs, and a modified $F^N_{\Delta H}$ limit equation for Indian Point Unit 3 and find them acceptable from a fuel mechanical design, nuclear, and thermal-hydraulic standpoint. These changes are identical to those approved for previous Westinghouse reload cores. In addition we have reviewed the proposed changes to the Technical Specifications to allow 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%. This request was supported by the Westinghouse analyses, WCAP-10704 and WCAP-10705 as discussed in section 3.0 of this evaluation.

3.0 STEAM GENERATOR ASYMMETRIC VALVE PLUGGING

The staff has reviewed the effect of up to 30% tube plugging in one steam generator on reactivity initiated transients and accidents as presented in WCAP-10704 (Proprietary) and WCAP-10705 (Non-Proprietary), "Safety Evaluation for Indian Point Unit 3 With Asymmetric Tube Plugging Among Steam Generators", dated October 1984. The tube plugging distribution analyzed was 30% in one steam generator, two steam generators at 24% and the fourth steam generator at 8%, resulting in an equivalent average plugging level in all steam generators of 24%. Safety analyses supporting operation with tube plugging of up to 24% in any steam generator and a 24% uniform tube plugging has previously been reviewed and approved by the NRC.

3.1 REACTIVITY INITIATED TRANSIENTS AND ACCIDENTS

The control rod withdrawal event from a subcritical condition (FSAR Section 14.1.1), the dropped control bank event (FSAR Section 14.1.1), and the control rod ejection accident (FSAR Section 14.2.6) were not reanalyzed. A reevaluation of these events has shown that they remain bounded by the 24% uniform plugging reference analysis cases, provided that the reference analysis DNB margins and operating limits are maintained. These requirements are ensured by maintaining conservative reactor coolant flows (323,600 gpm) and conservative vessel inlet temperature (546.9°F) relative to the 24% uniform tube plugging analyses.

The control rod withdrawal event from a power condition (FSAR Section 14.1.2) was reanalyzed since core DNB protection for this event is provided by the overtemperature delta-T trip. Because of the asymmetric tube plugging assumptions, inlet temperature will vary from loop to loop possibly affecting the delta-T protection system. The results indicate that the minimum DNBR remains above the 1.30 limit for all achievable reactivity insertion rates.

The dropped control rod event (FSAR Section 14.1.3) was also reanalyzed since it may cause a reactor trip on overtemperature delta-T. A DNB evaluation at the limiting condition in the transient shows that the DNBR remains above the 1.30 allowable limit.

To account for asymmetric tube plugging, two adjustments are required in the control and protection system instrumentation. In order to preserve core DNB margin during steady-state operation, the average vessel temperature (T_{avg}) program in the reactor control system must be adjusted such that no cold leg temperature exceeds its allowable value. Also, in order to prevent loop temperature asymmetries from causing core safety limits to be exceeded, the overtemperature delta-T reactor trip channels must be calibrated during power operation in terms of both the delta-T and T_{avg} indicated by each channel at nominal full power.

Technical Specification 3.1.A.6 and its associated Basis has been added to account for a maximum indicated T_{avg} of 576°F, which includes a 2°F measurement uncertainty. The licensee has shown that a maximum indicated T_{avg} of 576.7°F is calculated for a full power inlet temperature (T_{cold}) of 546.9°F at a flow of 323,600 gpm. The analyses in this report (WCAP-10704) assumed an inlet temperature of 546.9°F, which includes a 2°F allowance for control deadband and a 2°F allowance for temperature error. The Technical Specification T_{avg} limit of 576°F is, therefore, slightly conservative relative to the value used in the asymmetric tube plugging analyses and is acceptable.

The core safety limits are preserved, in part, by the overtemperature delta-T and overpower delta-T reactor trips. As mentioned earlier, asymmetric tube plugging may cause asymmetries in measured loop temperatures. However, with the proposed modifications to Technical Specifications 2.3.1.B.(4) and 2.3.1.B.(5), these asymmetries should not cause significant loop-to-loop variations in the delta-T protection system because each channel would now be calibrated at power based on measured temperatures in that loop. Therefore, the effect of loop average temperature asymmetry would be factored

into the overtemperature delta-T equation, effectively making the margin to trip equal for all channels and independent of temperature asymmetries. We, therefore, find these modifications to be acceptable.

We have reviewed the impact of the tube plugging asymmetry on the Indian Point Unit 3 FSAR Chapter 14 analyses of the reactivity initiated events (i.e., control rod withdrawal, dropped control bank, dropped control rod, and control rod ejection) and find that they remain bounded by the existing 24% uniform tube plugging analysis previously approved.

The steps that must be taken to ensure there will be no safety criteria violations are the following:

1. The reactor vessel flow must be equal to or greater than 323,600 gpm.

2. During steady-state operation at full power, the hottest cold leg inlet temperature must not exceed 542.9°F plus 2°F for control deadband.

3. The overtemperature delta-T reactor trip channels must be calibrated during power operation in terms of both delta-T and T_{avg} indicated by each channel at nominal full power.

These steps have been satisfactorily incorporated into the proposed Technical Specifications.

3.2 REACTOR SYSTEMS FLUID TRANSIENTS AND ACCIDENTS

Analyses of non LOCA events were performed using the Westinghouse LOFTRAN code which has been generically approved by the NRC staff for reactor transient and accident evaluations. The following events were analyzed. Main steam line

break, loss of forced coolant flow, locked rotor, and loss of normal feedwater flow. With the exception of the locked rotor accident the reactor system pressure was calculated to remain below acceptable values and the minimum fuel DNBR was calculated to remain above 1.30.

For the locked rotor accident the margin to film boiling was calculated to fall below the 1.3 DNBR criteria which has been utilized in previous licensing calculations as the minimum condition for fuel failure. The licensee's evaluation in WCAP-10704 concluded that the fuel would not fail based on the temperature-time criteria of NUREG-0562 "Fuel Rod Failures as a Consequence of Departure from Nucleate Boiling or Dryout." The report was developed for the Office of Nuclear Reactor Research and has not been applied in licensing. It has been and continues to be the NRC position that cladding failure is assumed to occur when the fuel rod DNBR is less than the safety limit. This position provides conservatism to cover analytical uncertainties in the core thermal hydraulics, geometry and power peaking in additional to uncertainties in experimental accuracy.

The licensee was therefore requested to perform calculations for the locked rotor accident which assume that all fuel which experiences a DNBR of less than 1.3 fails. Loss of offsite power was requested to be assumed as required by GDC-17.

The licensee provided a reanalysis of the event which determined that 1.9% of the fuel rods would experience a DNBR below 1.3 and were assumed to fail. Other assumptions were made similar to the locked rotor analysis in the FSAR which was accepted by the staff in the original licensing basis.

Below a reactor power of 10% the P-7 control interlock does not permit reactor trip on low reactor system flow. The staff therefore requested analysis of loss of flow transients and accidents at 10% power without reactor trip which include the added resistance from tube plugging. The licensee provided results from a previous natural circulation analysis at 10% power which included conservative assumptions for primary system flow resistance including a locked reactor coolant pump rotor in each loop. Reactor system flow was calculated to be reduced to 6.8% of design. The fuel rods were calculated maintain a minimum DNBR in excess of 1.3 even considering the large power peaking factors allowed by the Technical Specifications at low power levels.

Detailed analyses were not preformed for events involving boron dilution, inactive loop startup, load swings or main feedwater overcooling. Instead the licensee evaluated these events and determined that asymmetric tube plugging would not have any significant consequence in excess of those previously evaluated and approved for uniform plugging. The staff agrees with this conclusion.

The licensee did not evaluate the consequences of feedwater line break accidents since this event was not included in the original design basis. The consequences from feedwater line breaks have been shown to be acceptable for similar Westinghouse plants. The heat load imbalance from asymmetric tube plugging would not have a major impact on the consequences from this event. The staff acceptance criteria in the Standard Review Plan permits calculated reactor systems pressures up to 120% of design for feedwater line breaks. This

Loss of offsite power was demonstrated to not be significant in the locked rotor analyses for determining the number of fuel rods in DNB since reactor trip would not perturb the electric grid such that offsite power might be lost during the time when DNBR went through its minimum value. The offsite dose consequences were evaluated assuming loss of offsite power so that steam release from the secondary system was assumed to enter the atmosphere directly through relief and safety valves rather than being relieved to the condenser. Steam flow to the atmosphere was calculated to be 5.07×10^5 lbs in 2 hours and the total steam release was calculated to be 9.92×10^5 lbs. Maximum Technical Specification steam generator tube leakage was assumed. Both the licensee and the staff determined that the offsite dose consequences would not exceed the 10 CFR 100 guidelines and therefore the consequences would be acceptable for a design basis accident.

Below a reactor power of 10% the P-7 control interlock does not permit reactor trip on low reactor system flow. The staff therefore requested analysis of loss of flow transients and accidents at 10% power without reactor trip which include the added resistance from tube plugging. The licensee provided results from a previous natural circulation analysis at 10% power which included conservative assumptions for primary system flow resistance including a locked reactor coolant pump rotor in each loop. Reactor system flow was calculated to be reduced to 6.8% of design. The fuel rods were calculated maintain a minimum DNBR in excess of 1.3 even considering the large power peaking factors allowed by the Technical Specifications at low power levels.

Detailed analyses were not preformed for events involving boron dilution, inactive loop startup, load swings or main feedwater overcooling. Instead the

licensee evaluated these events and determined that asymmetric tube plugging would not have any significant consequence in excess of those previously evaluated and approved for uniform plugging. The staff agrees with this conclusion.

The licensee did not evaluate the consequences of feedwater line break accidents since this event was not included in the original design basis. The consequences from feedwater line breaks have been shown to be acceptable for similar Westinghouse plants. The heat load imbalance from asymmetric tube plugging would not have a major impact on the consequences from this event. The staff acceptance criteria in the Standard Review Plan permits calculated reactor systems pressures up to 120% of design for feedwater line breaks. This is considerably greater than calculated for the most severe feedwater line break for similar Westinghouse designed plants and the staff concludes that excessive reactor system pressure will not occur at IP-3 as a result of feedwater line break even following asymmetric tube plugging.

The licensee has previously demonstrated that the consequences of both small and large break loss of coolant accidents would be acceptable with 24% uniform tube plugging. For large break LOCA tube plugging affects the results by increasing the resistance to steam flow during the reflood period which increases the time required to recover the core. Only about 30% of the total resistance to steam flow would be from the steam generators with 24% uniform plugging. With up to 30% of the tubes plugged in one steam generator, 24% plugging or less would exist in the other loops since the average plugging limit

will remain at 24%. The difference in the net resistance to steam flow between uniform plugging and asymmetric plugging cases is insignificant so that the staff concludes that the consequences from large break LOCA would not be changed from that previously approved by the staff for uniform plugging.

For small break LOCA tube plugging is of less significance than for large break LOCA. Very little steam flow is calculated to occur within the coolant loops. For the small break sizes producing the highest cladding temperatures, the steam generators are not relied upon for reactor system heat removal.

We conclude that Indian Point Unit 3 can be safely operated with a maximum asymmetric tube plugging distribution of 30% in one steam generator, as long as the average plugging remains at 24%.

Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria

for categorical exclusion set forth in 10 CFR Sec 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 this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: August 27, 1985

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