ATTACHMENT I
Proposed Technical Specifications Changes

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

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## 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:

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

Where P is the fraction of Rated Thermal Power.

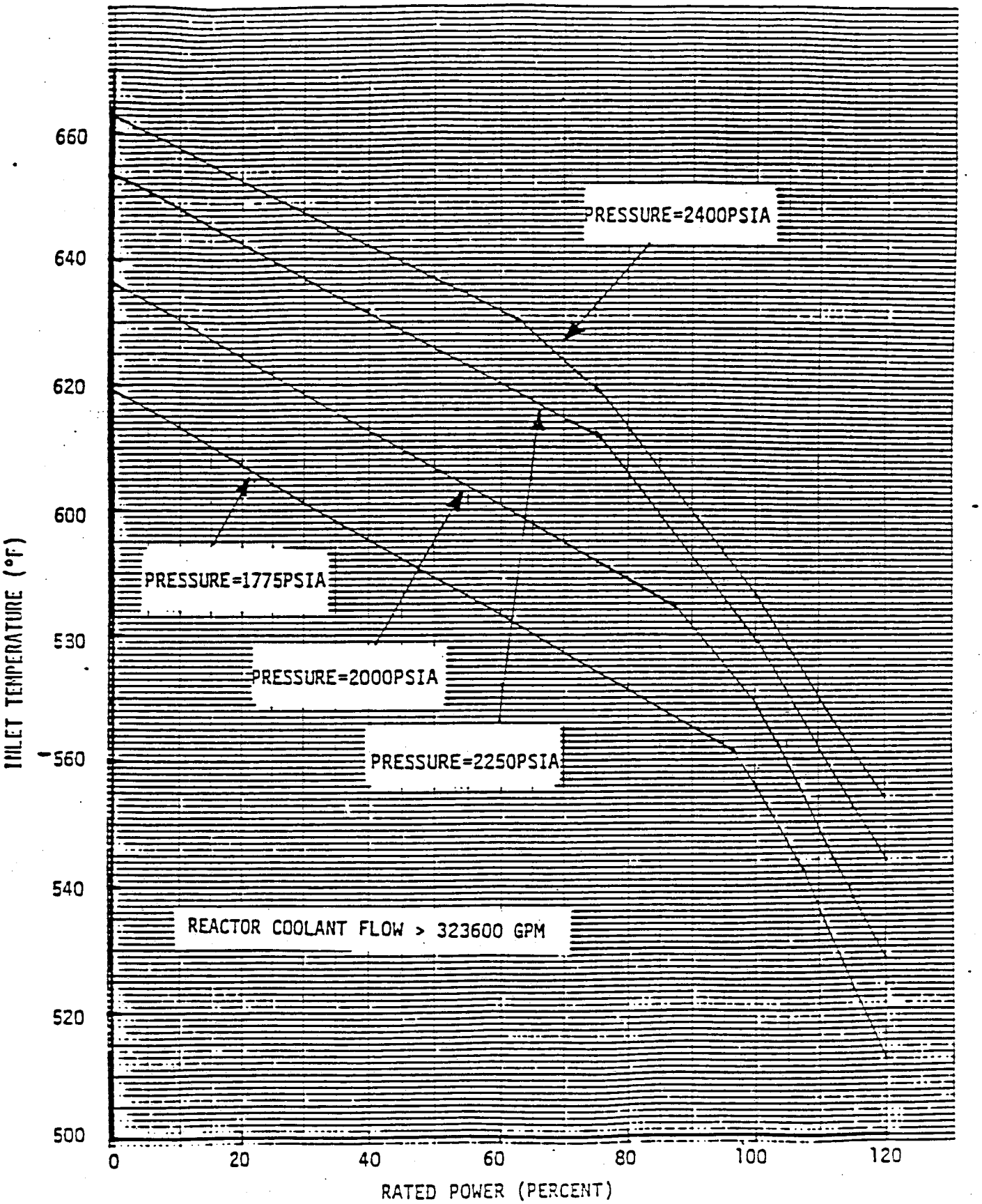
These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit (Figure 3.10-4) assuming the axial power imbalance is within the limits of the $f(\Delta I)$ function of the Overtemperature $\Delta T$ trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature $\Delta T$ trips will reduce the setpoints to provide protection consistent with core safety limits.

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. 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

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$\Delta T_o$ = Indicated $\Delta T$ 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 $\Delta T$ trip margin during steady state operation if the temperature, pressure and f ($\Delta I$) terms are zero.

$K_2$ is a constant which defines the dependence of the overtemperature $\Delta T$ set point to $T_{avg}$

$K_3$ is a constant which defines the dependence of the overtemperature $\Delta T$ set point to pressurizer pressure.

$\Delta I$ = $q_t - q_b$, 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 $\Delta T$ 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 $\Delta T$ trip setpoint shall be automatically reduced by an equivalent of 2.1 percent of rated power.

(5) Overpower $\Delta T$

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

where

| | |
|---|---|
| $\Delta T_o$ | =indicated $\Delta T$ 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 |
| $K_4$ | $\leq 1.089$ |
| $K_5$ | = 0 for decreasing average temperature<br>$\geq 0.175$ sec/°F for increasing average temperature |
| $K_6$ | = 0 for $T \leq T'$<br>$\geq 0.00116$ for $T > T'$ |
| $K_4$ | is a constant which defines the overpower $\Delta T$ trip margin during steady state operation if the temperature and the f ($\Delta I$) 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 $\Delta T$ 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 Fuel 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 ( $\Delta I$) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = $\Delta I$/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.

REFERENCE

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 , 1985, entitled "Proposed Technical Specifications Regarding the Cycle 4/5 Refueling".

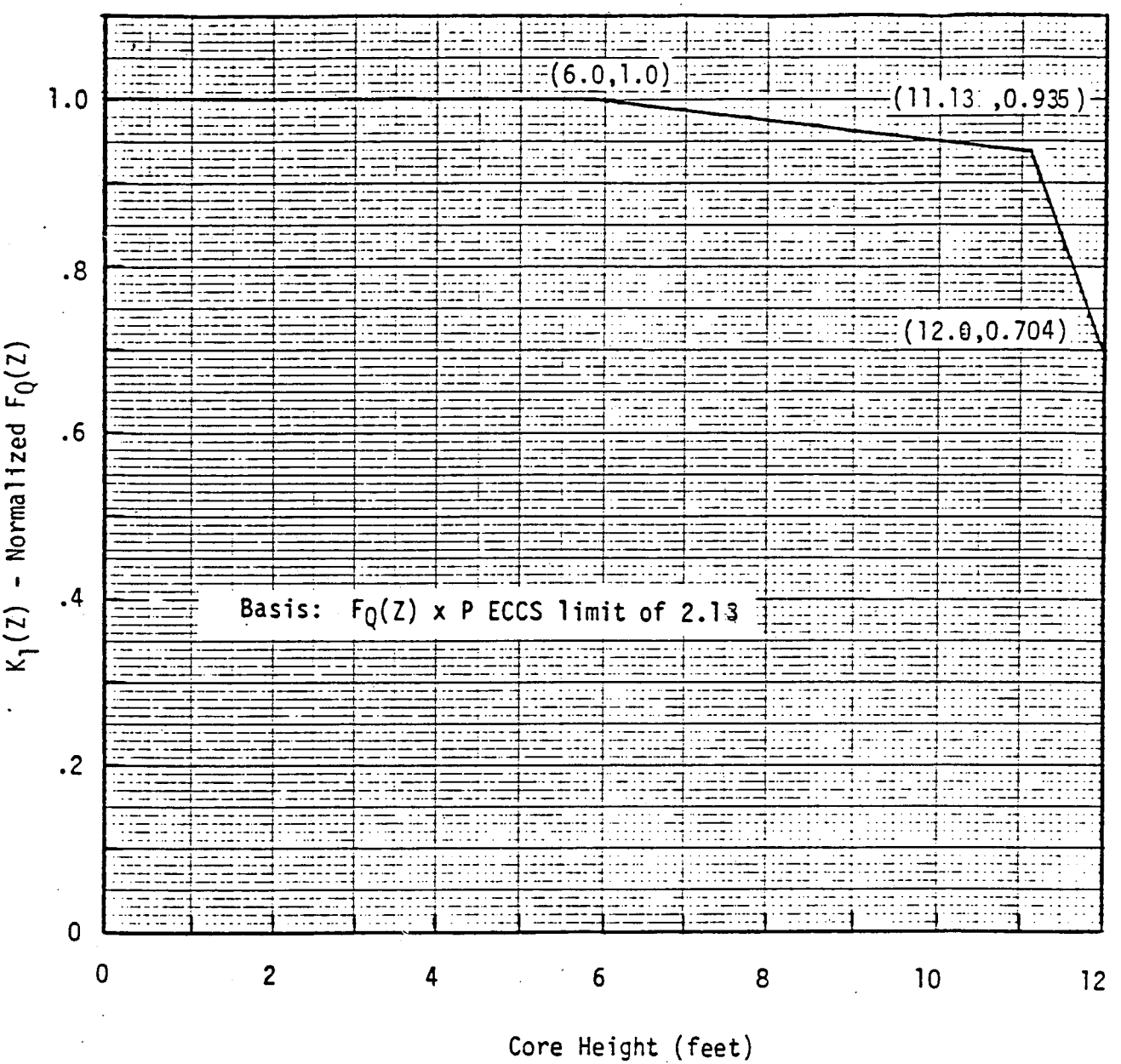



FIGURE 3.10-2 Hot Channel Factor Normalized Operating Evenlope

## 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.

ATTACHMENT II
Safety Evaluation of Proposed Technical Specifications

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

## 1.0 Introduction

Indian Point Unit 3 has been operating with all Westinghouse 15x15 low-parasitic (LOPAR) fueled cores. For Cycle 5 and subsequent cycles, it is planned to refuel with 15x15 optimized fuel assembly (OFA) regions supplied by the Westinghouse Electric Corporation. As a result, future core loadings would range from approximately a 40% OFA/60% LOPAR mixed core to eventually an all OFA fueled core. The OFA fuel will decrease neutron parasitic capture and thereby attain more efficient fuel usage. The 15x15 OFA fuel has design features similar to 15x15 LOPAR fuel which has had satisfactory operating performance in a number of nuclear plants. The major design difference involves the use of 5 intermediate Zircaloy grids for the OFA fuel versus 5 intermediate Inconel grids for LOPAR fuel.

The 15x15 OFA grids have design features similar to the Westinghouse 17x17 OFA which has been generically approved by the NRC via the review of the 17x17 OFA Reference Core Report.(1) Operating experience has been obtained on six 17x17 and four 14x14 demonstration OFAs which contained Zircaloy intermediate grids.(2) Four 17x17 OFAs and two 17x17 OFAs were observed to be in good condition after achieving assembly average burnups of 36,000 - 40,000 MWD/MTU and 22,000 MWD/MTU respectively. The 14x14 OFAs were examined in 1983 after two cycles of operation and assembly average burnups of approximately 21,000 MWD/MTU. One assembly experienced several fuel rod clad failures due to excessive clad fretting at the bottom Inconel grid location. This was caused by non-standard tooling methods used to manufacture only this failed assembly, and the failures are therefore considered an isolated event. The overall condition of the other three 14x14 demonstration assemblies was good with no fretting damage observed. Based on these results for the 17x17 and 14x14 demonstration assemblies, there is reason to expect good performance from the 15x15 OFA design.

Irradiation of the first 15x15 OFA reloads were initiated in the Cook Unit 1, Turkey Point Unit 3 and Zion Unit 1 plants during late 1983 and January 1984. Visual surveillance observations of the 15x15 OFA mechanical integrity during refueling should be available in early 1985, prior to the Indian Point Unit 3 Cycle 5 planned startup. These surveillance results are expected to confirm the good performance of 15x15 OFAs predicted by the OFA demonstration program.

The Westinghouse Wet Annular Burnable Absorber (WABA) rods will be a new feature introduced with the OFA fuel in order to enhance fuel cycle economy. WABA usage has been generically approved by an NRC evaluation which is incorporated into the approved version of the Westinghouse WABA Evaluation Report(3). New thimble plugs will be used to accommodate a reduced OFA thimble tube diameter. A minor design change to the OFA fuel pellets will be introduced via a small chamfer around the outer cylindrical surface of the pellet ends.

## 2.0 Summary and Conclusions

The Westinghouse Reload Safety Evaluation Methodology(4) was used to evaluate the transition from 15x15 LOPAR fuel to 15x15 OFAs for Indian Point Unit 3. Parameters were chosen to maximize the applicability of the transition evaluations for each reload cycle and to facilitate the safety evaluation of future reload cores. The mechanical, thermal and hydraulic, nuclear, and accident evaluations considered the transition core effects described for mixed cores in Chapter 18 of WCAP-9500(1). The summary of these evaluations for the Indian Point Unit 3 transitions to an all OFA core are given in the following sections of this submittal.

The transition design and safety evaluations consider the following nominal conditions: 3025 MWt core power, 2250 psia system pressure, 542.9°F core inlet temperature, and 323,600 gpm RCS thermal design flow. These conditions are consistent with the evaluations in Section 6.0 and supporting analysis of Reference 5, which justify safe operation

with up to 30% tube plugging in any steam generator provided a 24% plugging average for all steam generators is not exceeded. These conditions are also consistent with NRC approval[6] to operate with a maximum 24% tube plugging in any steam generator.

The results of evaluation/analyses and tests discussed in this report lead to the following conclusions:

2.1 The Westinghouse OFAs are mechanically and hydraulically compatible with the LOPAR fuel assemblies, control rods, and reactor internals interfaces.

2.2 Demonstration experience with 17x17 and 14x14 OFAs containing Zircaloy grids provides reason to expect satisfactory operation from 15x15 OFA Zircaloy grids.

2.3 As shown in the WABA evaluation report[3], the WABA rods/assemblies are compatible with the OFA and LOPAR fuel assemblies. The WABA rods satisfy all performance requirements for their design life.

2.4 Changes in the nuclear characteristics due to the transition from LOPAR to OFA fuel will be within the range normally seen from cycle-to-cycle due to fuel management effects.

2.5 Plant operating limitations given in the Technical Specifications will be satisfied with the proposed changes noted in Section 7.0 of this report. These are applicable to cores containing any combination of Westinghouse 15x15 OFAs and LOPAR assemblies, including a full-OFA core. Also, plant operating limitations will be satisfied for tube plugging levels up to 30% in any steam generator provide the average for all steam generators does not exceed 24% plugging.

2.6 The impact of using chamfered pellets in reload fuel for Indian Point Unit 3 has been evaluated. Results of the evaluation show that all core design criteria and safety limits (including LOCA and non-LOCA transients) are satisfied when using chamfered or unchamfered pellets.

2.7 A reference is established upon which to base future cycle safety evaluations.

## 3.0 Mechanical Evaluation

The mechanical design requirements and criteria for the 17x17 OFA design are described in WCAP-9500 which was approved by the NRC[1]. The 15x15 OFA design meets these same basic requirements and criteria except for a change in the fuel assembly nonoperational 6g loading design basis given in Section 4.2.1.5.1.a of WCAP-9500. The 6g value is changed to "4g axial and 6g lateral loading with dimensional stability". Notification of Westinghouse's plans to generically incorporate this change and the justification of no unreviewed safety question was previously transmitted to the NRC via Reference 7.

The OFAs have been designed to be compatible with the LOPAR assemblies, reactor internals interfaces, and fuel handling and refueling equipment. Figure 1 presents a comparison of the OFA and LOPAR fuel assembly. The grid elevations for the two assembly designs match, minimizing mechanical and hydraulic interaction. The assembly envelopes, fuel rod design/dimensions, and the top and bottom Inconel grids are the same. The ends of each OFA fuel pellet have a small chamfer at the outer cylindrical surface, whereas unchamfered pellets are in LOPAR assemblies. Also, a smaller fuel rod plenum spring can be used for OFAs; the spring force would be sufficient to satisfy the new 4g axial and 6g lateral load basis.

The five intermediate 15x15 OFA Zircaloy grids have thicker and wider straps than the 15x15 LOPAR Inconel grids to compensate for the difference in material strength properties. Impact tests that have been

performed at 600°F to obtain the dynamic strength data verify that the Zircaloy grid strength data at reactor operating conditions is structurally acceptable. The 15x15 OFA Zircaloy grid design has approximately 7 percent less crush strength than the 15x15 Inconel grid design.

The 15x15 OFA guide thimbles are similar in design to their counterparts in the LOPAR fuel assemblies except for 13 mil ID and OD reduction in the guide thimble above the dashpot. The reduction in OFA thimble diameter is due to the use of the thicker Zircaloy grid straps. The new design continues to provide an adequate nominal diametral clearance for control rods as well as other core components. Due to the reduced clearance, the rod drop time to the dashpot for accident analyses has been conservatively increased from 1.8 seconds for the LOPAR assembly to 2.4 seconds for the OFA. The increase in rod drop time required accident reanalyses as described in Section 6.0.

The OFA has about a 4.5 percent increase in hydraulic resistance to flow compared to a LOPAR assembly, primarily due to the thicker and wider OFA Zircaloy grid straps. This results in an increased OFA lift force and requires the use of 3-leaf holddown springs in the top nozzle instead of the LOPAR assembly 2-leaf springs (see Figure 1). The 3-leaf spring has the same height and provides additional holddown force margin when compared to the LOPAR 2-leaf spring. The 3-leaf spring design has been successfully used in the 17x17 OFA demonstration program and other 15x15 LOPAR assemblies. The change to the 3-leaf spring is fully compatible with the LOPAR assembly and the handling tools at the Indian Point plant. The 15x15 OFA bottom nozzle has similar design features as the LOPAR assembly bottom nozzle. The OFA bottom nozzle design has a reconstitutable feature which allows it to be easily removed. A locking cup is used to lock the thimble screw of a guide thimble tube in place, instead of the lockwire as used for the standard LOPAR nozzle design. The reconstitutable nozzle design facilitates remote removal of the bottom nozzle and relocking of thimble screws as the bottom nozzle is reattached.

The rod bow magnitude for the optimized fuel assemblies is predicted to be less than that of the 15x15 LOPAR assemblies. The optimized fuel assembly will have reduced grid forces (due to Zircaloy grid) and the same fuel rod dimensions as the LOPAR assembly fuel rod, which should tend to decrease OFA rod bow compared to LOPAR fuel. For a given burnup, the magnitude of rod bow for the optimized fuel assembly is conservatively taken to be the same as that applied to the 15x15 LOPAR fuel assembly. The impact of rod bow in DNBR penalties is discussed in Section 5.0.

The wear of fuel rod cladding is dependent on both the support provided by the assembly skeleton and the flow environment to which it is subjected. Hydraulic flow tests, as described in Section 5.0, were performed to verify the compatibility of the 15x15 OFA and LOPAR designs. The results of the tests show that no significant OFA or LOPAR fuel rod clad wear occurs due to the small amount of crossflow between fuel assemblies. Extrapolation of test results shows that fuel rod wear would be less than ten (10) percent of the cladding thickness for at least 48 months of reactor operation. This assures that clad wear will not impair fuel rod integrity.

## 4.0 Nuclear Evaluation

The transition from LOPAR to OFA fuel will not result in changes from the current nuclear design bases given in the FSAR[8] and applied to the cycle specific Reload Safety Evaluations (RSEs). Although the physics characteristics are slightly different for the OFA fuel when compared to LOPAR, evaluations show that differences are within the normal range of variations seen from cycle to cycle.

Changes in the nuclear characteristics in transitions from LOPAR to OFA cores will be primarily due to fuel management considerations (number of feed assemblies, feed enrichment, cycle burnup, etc.) and not due to the change in fuel assembly design. The standard calculational methods as described in Reference 4 continue to apply. As is current practice,

each reload core design will be evaluated to assure that design and safety limits are satisfied according to the reload methodology[4].

## 5.0 Thermal and Hydraulic Evaluation

The hydraulic effects of having a mixed OFA-LOPAR core were evaluated by performing hydraulic tests at the Westinghouse Fuel Assembly Test System (FATS) facility. A side-by-side OFA and LOPAR fuel assembly arrangement was tested under hydraulic flow conditions which approximate reactor conditions. Test results provided lift forces, pressure drops, cross-flow and fuel vibrations, and fuel rod clad wear. Based on these test results, it was concluded that hydraulic compatibility exists between OFA and LOPAR assemblies. The 15x15 OFA Hydraulic Flow Testing Report[9], provided for NRC review associated with the Zion OFA licensing submittal, applies also to the Indian Point Unit 3 hydraulic evaluation.

Calculational methods for the 15x15 OFA are the same as those currently used on the 15x15 LOPAR fuel assembly and described in the FSAR[8] and fuel densification document.[10] These methods are applicable to the evaluation of a core containing both 15x15 LOPAR and 15x15 OFAs, except for the application of the approved WRB-1 DNB correlation[11] for the OFAs. The present DNB safety evaluations for LOPAR fuel use the W-3 L-Grid DNB correlation with a design limit DNBR of 1.30. The evaluations contain a generic DNBR margin of 18.1 percent. Due to significant improvement in the accuracy of the critical heat flux prediction with the WRB-1 correlation[11] over previous DNB correlations, a DNBR limit of 1.17 is used in this application. In addition to the use of the WRB-1 correlation, a specific plant DNBR margin allowance has been considered in the Indian Point Unit 3 OFA analysis. In particular, the DNBR value of 1.43 was employed in safety analyses, resulting in 18.1 percent DNBR margin as defined in the following equation:

$$\frac{\text{DNBR LIMIT}}{\text{1-MARGIN}} = \frac{1.17}{1-0.181} = 1.43$$

In the transition cycles, the 18.1 percent margin is sufficient to cover the maximum 14.9 percent rod bow penalty at Loss of Flow Conditions[12] and a 3 percent transition core penalty. The 3 percent penalty was determined by analyzing 15x15 OFA and LOPAR assembly loading patterns at various core conditions in the same manner as for the 17x17 OFA/LOPAR fuel which was reviewed and approved by the NRC.[13] The 3 percent penalty is the same as that approved by the NRC for the Zion and Turkey Point OFA/LOPAR fuel mixed cores. When the full transition is complete (all LOPAR assemblies removed from core), the transition core penalty will no longer apply to OFA assemblies.

## 6.0 Accident Evaluation

Those accidents analyzed and reported in the FSAR[8] which could potentially be affected by the OFA reload have been reviewed. As discussed in Section 3.0, rod drop time (to dashpot) is increased from 1.8 to 2.4 seconds for control rods in OFA guide thimbles. Accident transients significantly affected are "fast" transients for which the protection system responds by tripping the reactor within a few seconds after the transient begins. The transients that fall into this category are Loss of Flow, Locked Rotor, and Rod Ejection. All other non-LOCA accidents analyzed in the FSAR have been evaluated. This evaluation showed that these accidents are not significantly affected by the increased rod drop time.

The simultaneous Loss of Flow from all four coolant pumps, Locked Rotor, and Rod Ejection accidents were reanalyzed to account for the increased rod drop time. Results for these accidents showed that all the safety limits and criteria are satisfied for the increased rod drop time. These accidents are bounding with respect to either assumption of 24% uniform steam generator tube plugging or asymmetric tube plugging[5].

The current Technical Specification core safety limits (T.S. Figure 2.1-1) are no longer applicable and should be replaced with the Figure 2.1-1 given in Attachment A of this submittal. New core limit curves and axial offset envelopes were calculated to accommodate an increased Nuclear Hot Channel Factor ($F^N_{\Delta H}$) at partial power based on a change from K = 0.2 to K = 0.3 in the following equation:
$F^N_{\Delta H} = 1.55\ [1 + K\ (1-P)]$, where P is the fraction of full power and the K multiplier is the partial power correction factor.

The safety analyses are not impacted by the K multiplier change, but Technical Specification changes are needed as shown in Section 7.0. An evaluation has determined that even with the increased rod drop time, the current Overpower ΔT and Overtemperature ΔT setpoint equation constants as given in the plant Technical Specifications are limiting and will be maintained. Therefore, no reanalysis is required for OTΔT trip events.

The difference in fuel assembly flow resistance ($K/A^2$) for the 15x15 LOPAR and 15x15 OFA designs impact the blowdown and reflood transients of a postulated large break LOCA. The 15x15 OFA increases the hydraulic resistance to flow by about 4.5 percent. Evaluations of hydraulic resistance increases has led to a conclusion that hydraulic resistance mismatches of less than 10 percent have an insignificant effect on blowdown cooling. Sensitivity studies have shown that for the reflood portion of the large LOCA transient, a flow reduction of five percent would lead to a PCT increase of 19°F. The hydraulic resistance increase for 15x15 OFA will cause approximately a 2.2 percent reduction in reflood flow rate for the 15x15 OFAs during the transition period. This will result in approximately a 10°F increase in PCT for the transition period. Once a full core of 15x15 OFA is achieved, the reflood penalty due to the increased core hydraulic resistance will be about .02 percent in flooding rate. Therefore the transition from LOPAR to OFA does not significantly affect the current LOCA analysis (ΔPCT<20°F).

However, since there is insufficient ΔPCT margin in the approved LOCA analysis to account to for the 10°F PCT penalty, the LOCA $F_Q$ will be

reduced by 0.01 during the transition period. Once a full core of 15x15 OFA has been installed, operation at the approved LOCA analysis $F_Q$ will be valid without accounting for a mixed core penalty of 10°F PCT or a 0.01 $\Delta F_Q$. These restrictions are not impacted by the asymmetry of the tube plugging.

Evaluations of the FSAR accidents have shown negligible changes in analyses results when the impact of using chamfered fuel pellets is considered. The current accident analyses of record remain applicable when using chamfered and/or unchamfered fuel pellets.

## 7.0 Technical Specification Changes

Based on the preceeding evaluations, the following technical specification changes for Indian Point Unit 3 are required to support the transition to OFA:

1. Modifications to Specification 3.10.8 to permit an increase in the shutdown and control rod drop time to 2.4 seconds.

2. Modifications to Specification 5.3.A.4 to permit the use of the WABA rods.

3. Modifications to Specification Bases 2.3 and 3.10 to incorporate the design limit DNBR for OFA.

4. Modifications to Specification 3.10.2.1 and Bases 2.1 to revise the $F_Q(Z)$ and $F^N_{\Delta H}$ limit equations. The slightly reduced 2.13 $F_Q$ limit (large LOCA break) accounts for a 0.01 $F_Q$ penalty during operation with OFA/LOPAR fueled mixed cores. The requested increase in the equations power multiplier from 0.2 to 0.3 has been previously reviewed and approved by the NRC for a number of other operating nuclear plants (e.g., Trojan, Zion, McGuire).

5. Revise Figure 3.10-2 to correspond to the reduced 2.13 $F_Q$ limit for OFA transition mixed cores.

6. Addition of a Specification 3.1.A.6 and addition to Specification Basis 3.1 to account for a maximum indicated $T_{avg}$ of 576°F to preserve the steady state DNBR margins.

7. Modifications to Specifications 2.3.1.B.(4) and 2.3.1.B.(5) to revise definitions to account for different individual loop temperatures as a result of asymmetric tube plugging evaluation (5) and to revise the $f_{\Delta H}^{N}(\Delta I)$ definitions(a), (b) and (c). The $f_{\Delta H}^{N}(\Delta I)$ revisions are made due to the change in the multiplier and new core limits for the OFA fuel transition.

8. Revised Figure 2.1-1 to incorporate new core limits for the OFA fuel transition.

9. Modifications to Specification 2.1 and its Basis to incorporate the design limit DNBR for OFA and to account for limitations of the asymmetric tube plugging evaluation(5).

The changes noted in items 1 to 9 above are given in the proposed Technical Specification page changes (see Attachment I of this submittal). In addition, information on three loop operation is deleted from Technical Specification Basis 2.3, since this operation is not permitted.

## 8.0 Significant Hazards Considerations Evaluation

Indian Point Unit 3 is currently operating in Cycle 4 with all Westinghouse 15x15 low parasitic (LOPAR) fuel assemblies. For Cycle 5 and subsequent cycles, it is proposed to refuel with 15x15 optimized fuel assembly (OFA) regions. Mechanical, nuclear, thermal-hydraulic, and accident evaluations have demonstrated full compatability of the OFA and LOPAR fuel assemblies in a mixed-fuel core during the transition to an all OFA core. These evaluations were performed utilizing methods previously reviewed and approved by the NRC staff.

Thermal-hydraulic calculational methods used in previous reload analyses are applicable to a mixed OFA/LOPAR fueled core, except for the use of the approved WRB-1 correlation to analyze the 15x15 OFAs. The effects of these changes on the safety analyses (such as a small increase in rod drop time) are minimal and are appropriately reflected in the proposed Technical Specifications. They do not create any new accident scenarios, produce more severe results or higher probabilities for existing accident scenarios, or result in significant reduction in safety margins.

The Authority considers that the proposed changes can be classified as not likely to involve significant hazard considerations since the proposed changes constitute a "change which may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan." (Example (VI), Federal Register, Vol. 48, No. 67 dated April 6, 1983, page 148701.

## 9.0 Impact of Change

This change will not impact the following:

- ALARA Program
- Fire Protection Program
- Emergency Plan
- FSAR or SER Conclusions
- Overall Plant Operations

## 10.0 Conclusion

The incorporation of these modifications: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the basis for any Technical Specification; d) does not constitute an unreviewed safety question as defined in 10 CFR 50.59; e) involves no significant hazards considerations as defined in 10 CFR 50.92.

## 11.0 References


1. Letter from R. L. Tedesco (NRC) to T. M. Anderson (Westinghouse), Safety Evaluation of WCAP-9500, "Reference Core Report - 17x17 Optimized Fuel Assembly," NRC SER letter dated May 22, 1981.

2. Skaritka, J., Iorii, J. A., "Operational Experience with Westinghouse Cores (up to December 31, 1983)," WCAP-8183, Rev. 13, September 1984.

3. E. P. Rahe (Westinghouse) letter to C. Thomas (NRC), Number NS-EPR-2871, December 30, 1983, Subject: Transmittal of Approved Versions WCAP-10121-P-A/10377-A, "Westinghouse Wet Annular Burnable Absorber Evaluation Report".

4. Bordelon, F. M., et. al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273 (Non-Prop.), March 1978.

5. Segletes, J., et. al., "Safety Evaluation of Indian Point Unit 3 with Asymmetric Tube Plugging Among Steam Generators", WCAP-10704 (Proprietary) and WCAP-10705 (Non-Proprietary), October 1984.

6. Letter from NRC to PASNY, Amendment 48 to Facility Operating License DPR-64, Docket 50-286, January 13, 1984.

7. Letter from E. P. Rahe, Jr. (Westinghouse) to L. E. Phillips (NRC) dated April 12, 1984, NS-EPR-2893, Subject: Fuel Handling Load Criteria (6g Versus 4g).

8. Indian Point Unit 3 Final Safety Analysis Report, Docket Number 50-286 Updated July 1984.

11.0 References (Cont'd)


9. Letter from F. G. Lentine (Commonwealth Edision) to H. R. Denton (NRC); Subject: Zion Station Units 1 and 2 Proposed Change to Facility Operating License Nos. DPR-39 and DAR-48, NRC Docket Nos. 50-295 and 50-304; November 4, 1983.

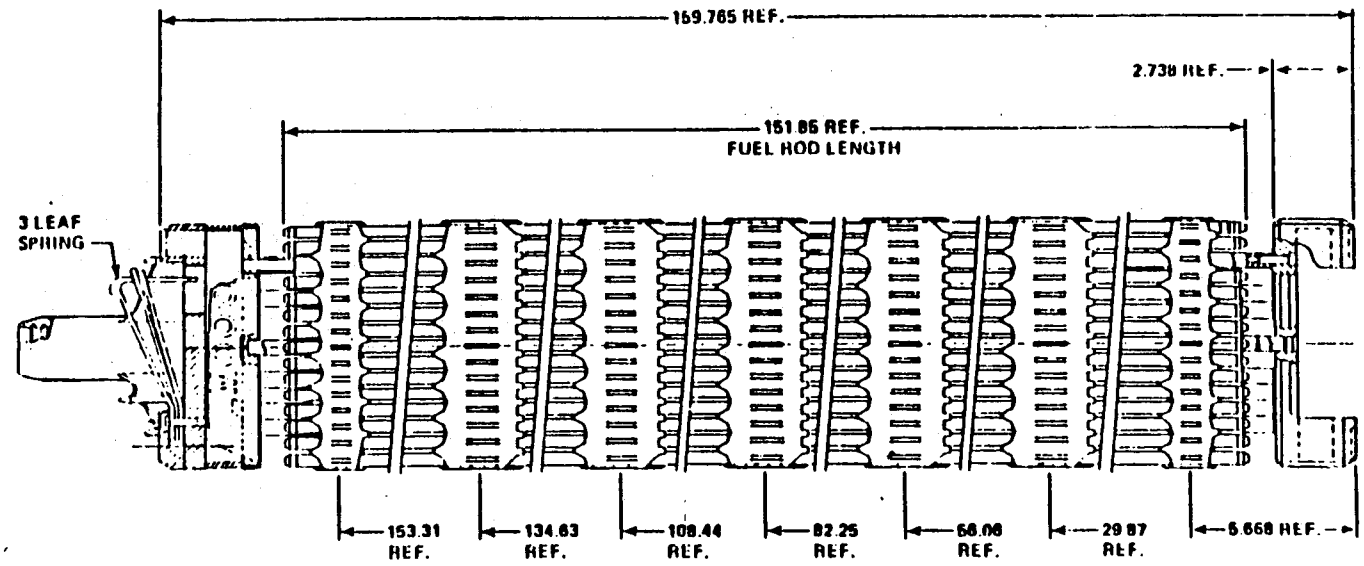
10. "Fuel Densification - Indian Point Unit 3" WCAP-8146, (Prop.), July 1973.

11. Motley, F. E., Hill, K. W., Cadek, F. F., Shefcheck, J., "New Westinghouse Correlation WRB-1 Predicting Critical Heat Flux in Rod Bundles with Mixing Grid, WCAP-8762-P-A (Prop.) and WCAPP-8763-A (Non-Prop.), July 1984.

12. J. F. Stolz, NRC letter to T. M. Anderson, (Westinghouse), "Staff Review of WCAP-8691," April 5, 1979.

13. Letter from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), Subject: Supplemental Acceptance Number 2 for Referencing of Licensing Topical Report WCAP-9500, NRC SER letter dated January 24, 1983.

15X15 OPTIMIZED FUEL ASSEMBLY




15X15 LOPAR FUEL ASSEMBLY

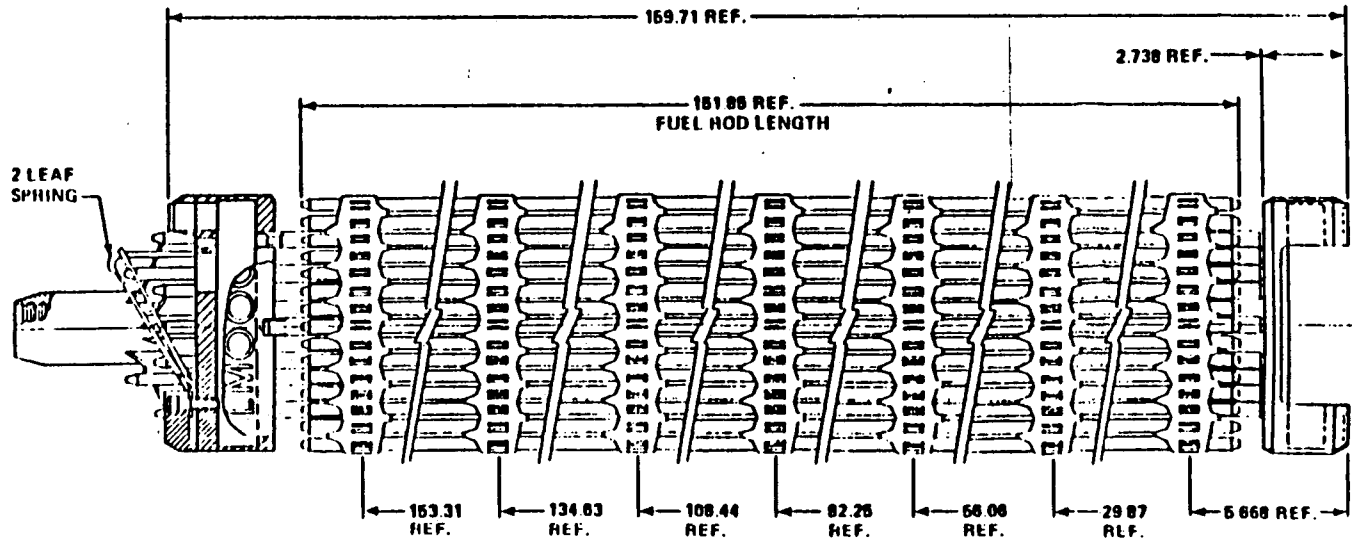



Figure 1