(3) Low pressurizer pressure - $\geq$1870 psig.

(4) Overtemperature ΔT

$$\Delta T \leq \Delta T_o \; [(K_1 - K_2 (T - T') + K_3 (P - P') - f(\Delta I)]$$

where: ΔT = Measured ΔT by hot and cold leg RTDs, °F
ΔT_o $\leq$ Indicated ΔT at rated power

T = Average temperature, °F
T' = Design full power Tave at rated power, $\leq$566.0°F
P = Pressurizer pressure, psig
P' = 2235 psig
K_1 $\leq$ 1.25
K_2 = 0.022
K_3 = 0.00095

and $f(\Delta I)$ is 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 such that:

(i) For $q_t - q_b$ between -36% and +7%, $f(\Delta I) = 0$, where q_t and q_b are percent RATED POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total POWER in percent of RATED POWER;

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -36%, the ΔT Trip Setpoint shall be automatically reduced by 2.14% of its value at RATED POWER; and

(iii) For each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT Trip Setpoint shall be automatically reduced by 2.15% of its value at RATED POWER.

(5) Overpower ΔT

$$\Delta T \leq \Delta T_o \; [K_4 - K_5 \frac{dT}{dt} - K_6 (T - T'')]$$

where: ΔT = Measured ΔT by hot and cold leg RTDs, °F

ΔT_o $\leq$ Indicated ΔT at rated power

T = Average temperature, °F

T'' = Indicated full power T_{avg} at rated power $\leq$ 566.0°F

K_4 $\leq$ 1.074

K_5 = Zero for decreasing average temperature

K_5 $\geq$ 0.188, for increasing average temperature (sec/°F)

K_6 $\geq$ 0.0015 for $T \geq T''$; $K_6 = 0$ for $T < T''$

$\frac{dT}{dt}$ = Rate of change of T_{avg}

G. REACTOR COOLANT SYSTEM PRESSURE, TEMPERATURE, AND FLOW RATE

Specifications

The following DNB related parameters pertain to four loop steady-state operation at power levels greater than 98% of rated full power:

a. Reactor Coolant System $T_{ave} \leq 573.5°F$

b. Pressurizer Pressure $\geq$ 2190 psig

c. Reactor Coolant System Total Flow Rate $\geq$ 331,840 gpm

Item (b), pressurizer pressure, is not applicable during either a thermal power change in excess of 5% of rated thermal power per minute, or a thermal power step change in excess of 10% of rated thermal power.

Under the applicable operating conditions, should reactor coolant temperature, T_{avg}, or pressurizer pressure exceed the values given in items (a) and (b), the parameter shall be restored to its applicable range within 2 hours.

Basis

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in a DNBR of less than the safety limit DNBRs.

The limits on reactor coolant system temperature, pressure and loop coolant flow represent those used in the accident analyses and are specified to assure that the values assumed in the accident analyses are not exceeded during steady-state four loop operation. Indicator uncertainties have not been accounted for in determining the DNB parameter limits on temperature and pressure.

Compliance with the specified ranges on reactor coolant system temperature and pressurizer pressure is demonstrated by verifying that the parameters are within their applicable ranges at least once each 12 hours.

Compliance with the specified range on Reactor Coolant System total flow rate is demonstrated by verifying the parameter is within it's range after each refueling cycle.

4.0 THERMAL AND HYDRAULIC EVALUATION

The Improved Thermal Design Procedure (ITDP) (Reference 3) and the THINC-IV (Reference 4 and 5) computer code are used for evaluation of both the standard and optimized fuel assemblies. The WRB-1 (Reference 6) DNB correlation is used in the 15x15 OFA analyses. The 15x15 LOPAR (STD) fuel analyses continue to use the W-3, L-grid (Reference 24) correlation.

The Departure from Nuclear Boiling Ratio (DNBR) correlation limits are 1.24 for the LOPAR fuel and 1.17 for the OFA. The thermal hydraulic design of this core is analyzed for operation at 3071.4 MWt core power which envelopes the current rated power of 2758 MWt. Thermal-Hydraulic Design Parameters are given in Table 4.1.

The design method employed to meet the DNB design basis is the Improved Thermal Design Procedure (ITDP) which has been approved by the NRC. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability with a 95 percent confidence level that the minimum DNBR will be greater than or equal to the correlation limit DNBR for the limiting power rod. Plant parameter uncertainties are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR limit, establishes a DNBR value which must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses for DNB related transients are performed using values of input parameters without uncertainties. For this application, the minimum required design DNBR values are 1.34 for OFA and 1.40 for LOPAR for thimble coldwall cells (three fuel rods and a thimble tube) and 1.35 for OFA and 1.47 for LOPAR for typical cell (four fuel rods).

In addition to the above considerations, specific plant DNBR margin has been considered in the analysis. In particular, the DNBR values of 1.47 and 1.52, for thimble and typical cells respectively, were employed in the safety analyses of the LOPAR fuel. A safety DNBR limit of 1.52 for both typical and thimble cells is used in design of the OFA. The DNBR margin between the DNBRs used in the safety analyses and the design DNBR values is broken down as

TABLE 4.1
Thermal and Hydraulic Design Parameters

ITDP Parameters		Design Value
Reactor Core Heat Output, Mwt		3071.4
	10^6 BTU/hr	10,483
Heat Generated in Fuel, %		97.4
Core Pressure, psia		2280
Radial Power Distribution	LOPAR	1.56[1+0.3(1-P)]
	OFA	1.59[1+0.3(1-P)]
Minimum DNBR at Nominal Conditions	Typical Flow Channel	2.41 LOPAR
		2.45 OFA
	Thimble Flow Channel	2.13 LOPAR
		2.33 OFA
Design Limit DNBR	Typical Flow Channel	1.47 LOPAR
		1.35 OFA
	Thimble Flow Channel	1.40 LOPAR
		1.34 OFA
Safety DNBR for Design	Typical Flow Channel	1.52 LOPAR
		1.52 OFA
	Thimble Flow Channel	1.47 LOPAR
		1.52 OFA
DNB Correlation	LOPAR	W-3, L-Grid
	OFA	WRB-1
HFP Nominal Coolant Conditions		
Vessel Minimum Measured Flow Rate (MMF) (Including Bypass),	10^6 lbm/hr	124.38
	GPM	330,000
Vessel Thermal Design Flow Rate (TDF) (Including Bypass)	10^6 lbm/hr	121.72
	GPM	322,800
Nominal Vessel/Core Inlet Temp, °F	based on TDF	547.7
	based on MMF	548.3
Vessel Average Temp, °F		579.7
Vessel Outlet Temp, °F	based on TDF	611.7
	based on MMF	611.1

g. Loss of External Electrical Load

h. Loss of Normal Feedwater

i. Reduction in Feedwater Enthalpy Incident

j. Excessive Load Increase Incident

k. Loss of All AC Power to the Station Auxiliaries

l. Rupture of a Steam Pipe

m. Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

All of the above Non-LOCA accidents which could potentially be affected by the OFA reload have been reviewed and include consideration for the following design and/or Technical Specification changes:

1. The control rod scram time to the dashpot is increased (as discussed in Section 2.2.2) from 1.8 seconds to 2.4 seconds. This increased drop time primarily affects the fast reactivity transients.

2. For events reanalyzed, the analyses were conservatively performed assuming a nominal thermal power level of 3071.4 MWt, a vessel average temperature range of 549.0°F to 579.7°F, and a thermal design flow rate of 322,800 gpm. These conditions will bound Cycle 10 operation at the current nominal thermal power level of 2758 MWt. These assumed conditions have the largest effect on transients that are limiting at full power conditions. In addition, fuel temperatures were calculated for the events reanalyzed based on the revised PAD computer code fuel thermal safety model (Reference 28).

3. As discussed in Section 4.0, the OFA transition includes implementation of the Improved Thermal Design Procedure (ITDP) using both the WRB-1 and W-3 L-grid DNB correlations for OFA and LOPAR fuels respectively, an increase in the nuclear enthalpy rise hot channel factor $F_{\Delta H}^{N}$

5. The core temperature is reduced and decay heat is removed for the extended period of time that is required to remove the heat from the long-lived radioactivity in the core.

The time sequence of events for all breaks analyzed is shown in Table 14.3-4 contained in Enclosure 2.

The large break LOCA analysis for Indian Point Unit 2, utilizing the BASH model, resulted in a peak clad temperature of 2039°F at a NSSS power level of 3083.4 Mwt for the limiting break case (C_D = 0.4 at a high vessel average temperature under minimum safeguards assumptions) at a total peaking factor of 2.32. The maximum local metal-water reaction was 5.54 percent, and the total metal-water reaction was less than 0.3 percent for all cases analyzed. The clad temperature turned around at a time when the core geometry is still amenable to cooling. Criteria 5 is addressed separately in a specific evaluation for each reload cycle.

The small impact of crossflow for transition core cycles is conservatively evaluated to be no greater than 10°F, which is easily accommodated in the margin to the 10CFR50.46 limits (i.e. transition PCT $\leq$ 2049°F).

It can be seen from the results of this large break ECCS analysis that Indian Point Unit 2 remains in compliance with the requirements of 10CFR50.46.

5.2.2 Small Break LOCA

5.2.2.1 Description of Analysis/Assumptions

The small break loss-of-coolant accident (LOCA) was analyzed using axial power shapes consistent with the peaking factor limits assumed for the transition cores at reactor core power of 3071.4 MWt thermal and 25% steam generator tube plugging. The NRC approved NOTRUMP small break ECCS Evaluation Model was employed to analyze a spectrum of cold leg break sizes (4 in., 6 in. and 8 in. equivalent diameter). Enclosure 2 contains a full description of the conditions and assumptions utilized for the small break LOCA analysis.

h. Loss of Normal Feedwater

i. Reduction in Feedwater Enthalpy Incident

j. Excessive Load Increase Incident

k. Loss of All AC Power to the Station Auxiliaries

l. Rupture of a Steam Pipe

m. Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

All of the above Non-LOCA accidents which could potentially be affected by the OFA reload have been reviewed and include consideration for the following design and/or Technical Specification changes:

1. The control rod scram time to the dashpot is increased (as discussed in Section 2.2.2) from 1.8 seconds to 2.4 seconds. This increased drop time primarily affects the fast reactivity transients.

2. For events reanalyzed, the analyses were conservatively performed assuming a nominal thermal power level of 3071.4 MWt, a vessel average temperature range of 549.0°F to 579.7°F, and a thermal design flow rate of 322,800 gpm. These conditions will bound Cycle 10 operation at the current nominal thermal power level of 2758 MWt. These assumed conditions have the largest effect on transients that are limiting at full power conditions. In addition, fuel temperatures were calculated for the events reanalyzed based on the revised PAD computer code fuel thermal safety model (Reference 3).

3. As discussed in Section 4.0, the OFA transition includes implementation of the Improved Thermal Design Procedure (ITDP) using both the WRB-1 and W-3 L-grid DNB correlations for OFA and LOPAR fuels respectively, an increase in the nuclear enthalpy rise hot channel factor $F^N_{\Delta H}$ limit from 1.55 to 1.62 and removal of all or some portion of the thimble tube plugging devices. A conservative set of core thermal safety limits, overtemperature

A.3.5.2 Method of Analysis

The analysis were performed, assuming implementation of the transition to OFA fuel, and a nominal core power of 3071.4 MWt which bounds the current nominal core power of 2758 MWt.

The method and assumptions used in the analysis are consistent with those employed in the FSAR, with the exception that the Improved Thermal Design Procedure was used for the Rods-In-DNB calculation.

The following effects of the Locked Rotor were investigated using the 2.4 second rod drop time:

1. Primary pressure transient.

2. Fuel clad temperature transient (This is calculated assuming film boiling in order to give the worst possible results).

3. DNB transient (for determining the percentage of rods in DNB for the offsite dose release calculations).

The following assumptions were used:

1. Initial operating conditions most adverse with respect to margin to clad temperature, RCS pressure;

   a. Power = 102% of nominal (1)
   b. Vessel Average Temperature = 587.2 °F (1)
   c. RCS Pressure = 2298.0 psia (2)

---

(1) For the RODS-IN-DNB calculation the nominal value was used according to the Improved Thermal Design Procedure.

(2) For the RODS-IN-DNB calculation the expected value of core pressure (2280 psia) was used according to the Improved Thermal Design Procedure.

2. Highest value (absolute) of Doppler Power coefficient and zero moderator temperature coefficient.

3. 4% ΔK trip reactivity from full power.

4. For clad temperature calculation DNB is assumed to occur at time = 0.

The flow coastdown transient was computed by the LOFTRAN code. The FACTRAN code was used to calculate fuel rod temperatures and heat flux distribution. The THINC code was used to calculate DNBR. The analysis was performed without offsite power available.

A.3.5.3 Results

Under the conditions used in the analysis, peak reactor coolant pressure was determined to be 2540 psia, and peak clad temperature 1676°F. Figures A.3-20 through A.3-23 show the core flow coastdown, nuclear power, reactor coolant pressure, and fuel clad temperature transients respectively. The sequence of events and summary of results is given in Table A.3-3.

The most limiting case yields no rods in DNB.

A.3.5.4 Conclusions

The 2.4 seconds rod drop time and other design changes associated with OFA can be accommodated by existing margins with regard to the Locked Rotor transient. The peak pressure of 2540 psia is below the maximum allowable value of 2750 psia and the peak clad temperature of 1676°F is well below the maximum (hot spot) average clad temperature limit of 2700°F. The safety criteria and dose release limits are not exceeded.

A.3.6 Rod Ejection

A.3.6.1 Introduction

This accident, which is reported in Section 14.2.6 of the FSAR, was reanalyzed to assure fuel rod enthalpy, melt and clad temperature criteria would not be violated by the increased rod drop time associated with OFA transition.

A.3.6.2 Method of Analysis

Methods and assumptions used in the analysis were consistent with those employed in the FSAR. The effects of rod ejection were investigated assuming the following.

1. A conservative value of trip rod worth is used assuming a stuck rod in addition to the ejected rod.

2. Initial Power = HZP or 1.02 x nominal HFP.

3. Initial Pressure = 2190.0 psia.

4. Initial Coolant Average Temperature = 587.2°F., HFP (547°F HZP)

5. Initial Fuel Temperature = 2400°F.

The Rod Ejection accident transient was simulated using the TWINKLE and FACTRAN computer codes. Four conditions were analyzed: EOL-HFP, EOL-HZP, BOL-HFP, BOL-HZP. Additional detailed information on the Rod Ejection accident is included in FSAR Section 14.2.6.

A.3.6.3 Results

The results are presented in Table A.3-4. Figures A.3-24 and A.3-25 show the nuclear power and fuel rod temperature transients for the EOL-HFP and EOL-HZP cases, respectively.

TABLE A.3-3

TIME SEQUENCE OF EVENTS

LOCKED ROTOR EVENT - HOT SPOT

Event	Time (Seconds)
Rotor in one pump seizes	0.
Reactor low flow trip point reached at	0.1
Rods begin to fall	1.1
Maximum RCS pressure occurs	3.0
Maximum clad temperature occurs	3.6

SUMMARY OF THE RESULTS

LOCKED ROTOR EVENT - HOT SPOT

Maximum Reactor Coolant System Pressure (psia) .... 2540.
Maximum Clad Avg Temperature (°F) ....1676.
Maximum Peak Fuel C/L Temp. (°F) ....3710.
% Zirconium Reacted ....0.2 %

TABLE A.3-4

SUMMARY OF ROD EJECTION ANALYSIS PARAMETERS AND RESULTS

Accident Parameters	Time in Cycle Beginning	Beginning	End	End
Initial Power, % Rated Power	0	102	0	102
Ejected Rod Worth, % Δk/k	.65	.17	.80	.20
Delayed Neutron Fraction (b_{eff})	.0050	.0050	0.0040	0.0040
FQ during Event	12.0	6.8	20.0	7.1
Results-Rod/Drop Time = 2.4 Secs				
Max. Fuel Centerline Temperature (°F)	2764	*	3606.	*
Max. Clad Average Temperature (°F)	1840	2175.	2525.	2124
Max. Fuel Enthalpy (Btu/lb)	174.2	304.2	247.5	296.9

*Less than 10% fuel centerline melt at fuel rod hot spot.

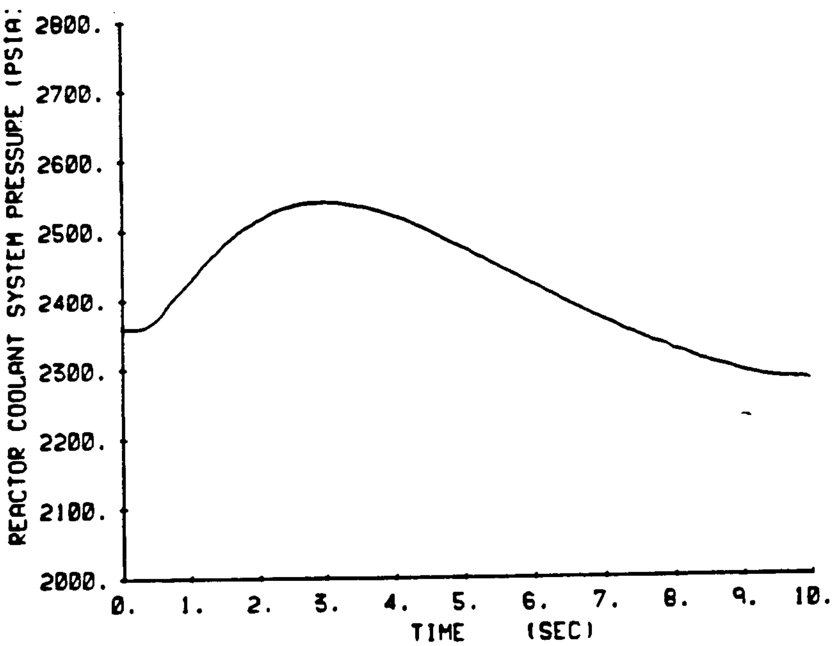

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FIGURE A.3-22

LOCKED ROTOR INCIDENT
RCS PRESSURE VS. TIME

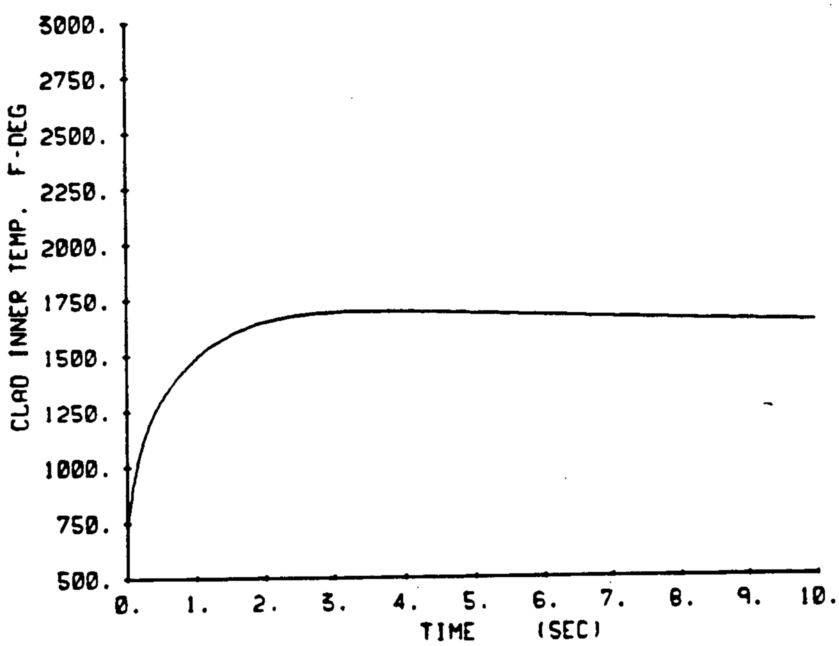

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FIGURE A.3-23

LOCKED ROTOR INCIDENT
CLAD INNER TEMPERATURE VS. TIME

7.0 BORON CONCENTRATION SHUTDOWN MARGIN

7.1 Description of Change

The proposed revision to Technical Specification 3.8.B.2 and the associated bases would decrease the required shutdown margin during refueling from 10% $\Delta k/k$ to 5% $\Delta k/k$ while fixing the minimum refueling boron concentration at 2000 ppm. To maintain consistency, a proposed change to Technical Specification 3.6.A.1 and its associated bases is also required to reflect the proposed change to the required shutdown margin during refueling given in Technical Specification 3.8.B.2. The proposed change in the minimum required shutdown margin during refueling allows for increased flexibility in fuel management and provides consistency with the required shutdown margin of the spent fuel pit given in Technical Specification section 5.4.

7.2 Safety Assessment

Whenever the reactor vessel head is less than fully tensioned, the plant operation must adhere to the Technical Specification 3.8 requirements for shutdown margin. The purpose of this requirement is to ensure that the reactor core is sufficiently subcritical during refueling such that adequate operator response time exists to mitigate the possible occurrence of an uncontrolled boron dilution transient as described in FSAR Section 14.1.5.2.1. The proposed change to Technical Specification 3.8.B.2 would require that the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following is met:

(1) a shutdown margin greater than or equal to 5% $\Delta k/k$, or,

(2) a boron concentration of greater than or equal to 2000 ppm.

A safety analysis for the boron dilution during refueling event as described in FSAR Section 14.1.5.2.1 has been performed based on the proposed change to Technical Specification 3.8.B.2. The results of this analysis have demonstrated conformance with the acceptable design and regulatory requirements assuming the proposed changes to Technical Specification 3.8.B.2.

The purpose of Technical Specification 3.6.A.1 regarding whenever the reactor vessel head is less than fully tensioned and the refueling shutdown margin requirement is not met is to ensure containment integrity is maintained until acceptable shutdown margin requirements are met for refueling. Since the proposed revision to Technical Specification 3.8.B.2 changes the refueling shutdown margin requirements, a proposed revision to Technical Specification 3.6.A.1 is necessary to maintain consistency. This proposed revision does not change the purpose of Technical Specification 3.6.A.1. and, therefore, is considered to be an administrative change and does not affect any margin to safety.

7.3 Basis for No Significant Hazards Consideration Determination

Consistent with the Commission's criteria in 10 CFR 50.92, we have determined that the proposed changes do not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision is supported by conservative analyses utilizing approved methodology. These analyses have demonstrated conformance to the applicable design and regulatory criteria.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change to the refueling shutdown margin and minimum boron concentration does not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require