ATTACHMENT I TO IPN-89-027

REVISED PAGES FOR APPENDIX B OF THE
PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS
REGARDING THE TRANSITION TO WESTINGHOUSE
15x15 VANTAGE 5 FUEL

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

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a. Power = 102% of nominal (1)

b. Average Coolant Temperature = 579.7 °F (1)

c. Initial RCS Pressure = 2290 psia (1)

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

3. 4% ΔK trip reactivity from full power.

4. For the clad temperature calculation film boiling is assumed to maximize clad temperatures.

The analysis was performed assuming offsite power available for 5 seconds after the time of shaft seizure/break. After 5 seconds, offsite power is assumed to be lost and flow coastdown of the unfaulted RCPs is initiated. 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.

B.3.6.3 Results

Peak reactor coolant pressure = 2566 psia and peak clad temperature = 1862 °F. The maximum zirconium-steam reaction at the core hot spot is 0.4% by weight. Figures B.3-30 through B.3-33 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 B.3-5.

The most limiting case yields no rods in DNB.

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

B.3.6.4 Conclusions

The increase in peaking factors and other design changes associated with V-5 can be accommodated by existing margins with regard to the Locked Rotor transient. The peak pressure of 2566 psia is below the maximum allowable value of 2750 psia and the peak clad temperature of 1862 °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.

TABLE B.3-5

TIME SEQUENCE OF EVENTS

LOCKED ROTOR EVENT - HOT SPOT

Event	Time (Seconds)
Rotor in one pump seizes	0.0
Reactor low flow trip point reached at	0.1
Rods begin to fall	1.1
Maximum RCS pressure occurs	3.6
Peak clad temperature occurs	3.7

SUMMARY OF THE RESULTS

LOCKED ROTOR EVENT - HOT SPOT

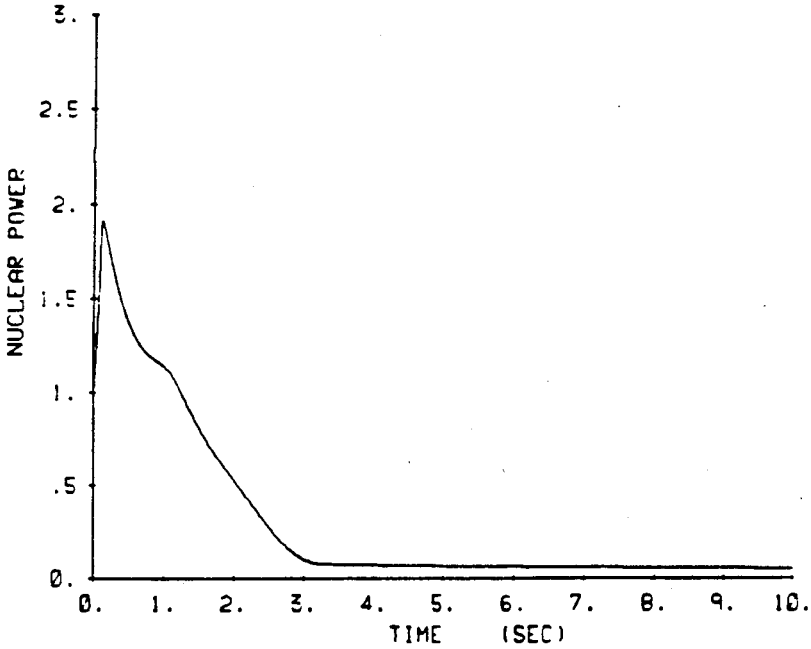
Maximum Reactor Coolant System Pressure (psia) .... 2566.
Peak Clad Average Temperature (° F) ..... 1862.
Peak Fuel Centerline Temperature (° F) ..... 3704.
% Zirconium Reacted ..... 0.4 %
Rod in DNB ..... 0.0 %

TABLE B.3-11

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, % $\Delta k/k$	.65	.17	.80	.20
Delayed Neutron Fraction (b_{eff})	.0050	.0050	0.0040	0.0040
F_Q during Event	12.0	6.8	20.0	7.1
Results				
Max. Fuel Centerline Temperature (°F)	2760	*	3544.	*
Max. Clad Average Temperature (°F)	1817	2143.	2436.	2092
Max. Fuel Enthalpy (Btu/lb)	173.4	313.7	241.2	304.1

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



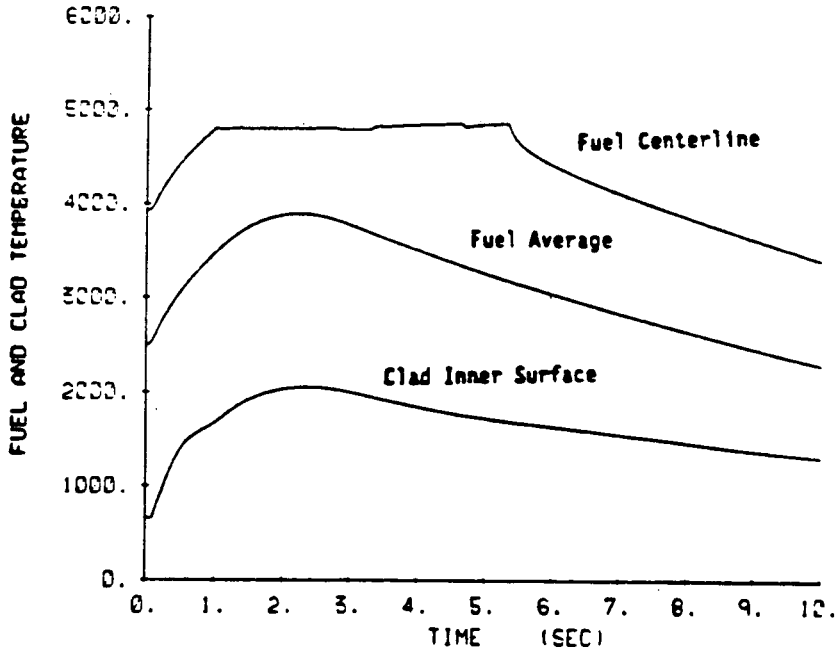


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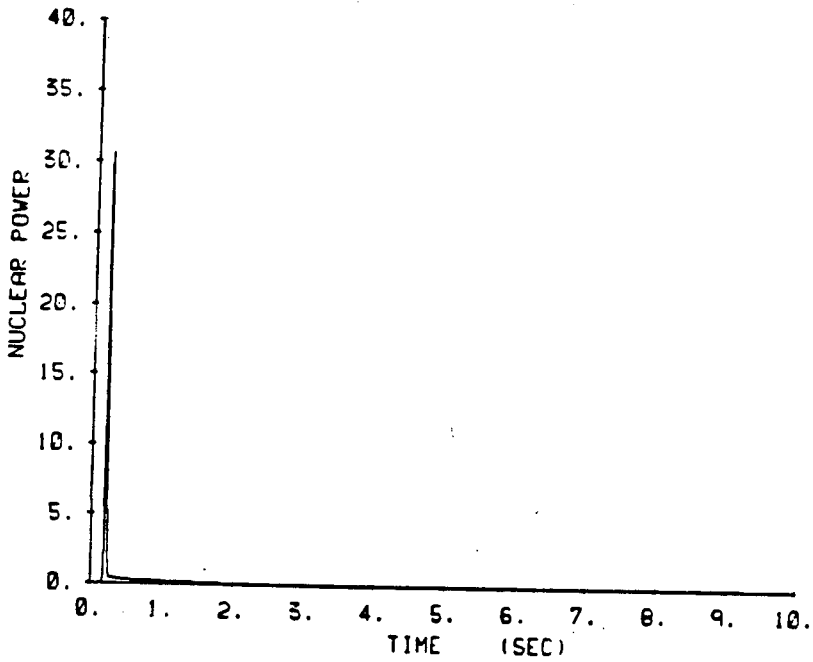
FIGURE B.3-67

ROD EJECTION / EOL-HFP
NUCLEAR POWER AND FUEL ROD
TEMPERATURE VS. TIME



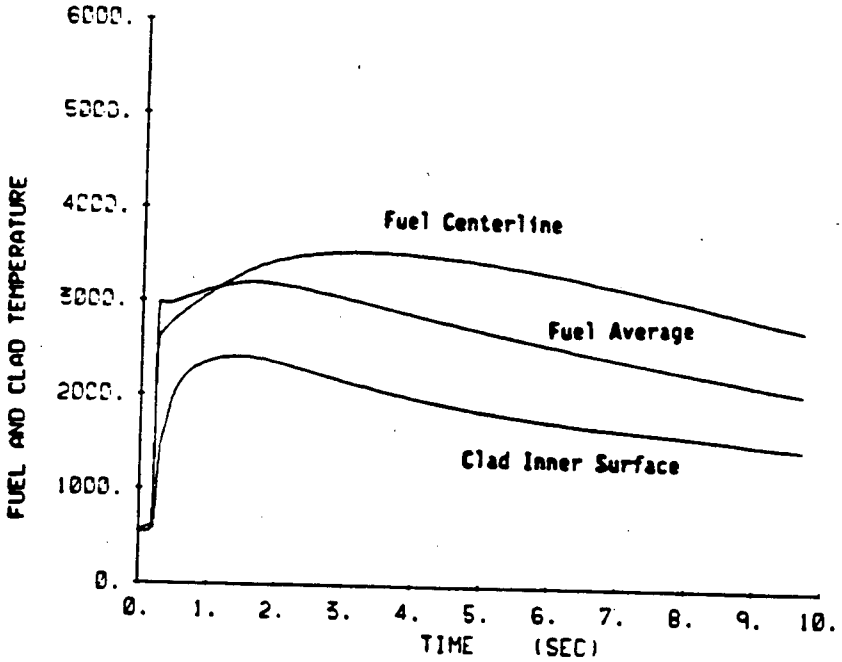


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FIGURE B3-68

ROD EJECTION / EOL-HZP
NUCLEAR POWER AND FUEL
ROD TEMPERATURE VS. TIME

ATTACHMENT II TO IPN-89-027

ADDITIONAL INFORMATION CONCERNING CONDITIONS 1, 2, 3 AND 5 OF THE GENERIC VANTAGES 5 NRC SAFETY EVALUATION REPORT

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
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DPR-64

CONDITION 1

The statistical convolution method described in WCAP-10125 for the evaluation of initial fuel rod to nozzle growth gap has not been approved. This method should not be used in Vantage 5.

RESPONSE

Worst case fabrication tolerances and fuel rod and assembly growth are used to determine the initial fuel rod to nozzle growth gaps in the evaluation of fuel rod performance summarized in Section 3 of Attachment 2 to Reference (1), as per Section M of References 5. This is in compliance with Condition 1 of the Vantage 5 NRC Safety Evaluation Report.

CONDITION 2

For each plant application, it must be demonstrated that the Loss of Coolant Accident (LOCA)/seismic loads considered in WCAP-9401 bound the plant in question; otherwise additional analysis will be required to demonstrate the fuel assembly structural integrity.

RESPONSE

The fuel assembly for IP3 is a 15x15 7-grid assembly with features of the Vantage 5 design. Structurally, the fuel assembly design is identical to the Optimized Fuel Assembly (OFA) fuel assembly. The analysis in WCAP 9401 is not applicable to IP3.

An evaluation of the optimized core and the mixed fuel assembly core for IP3 considering the combined lateral effects of a LOCA event and seismic event has been performed. The grid load results indicate that both the OFA and the Vantage 5 fuel assembly designs meet all Standard Review Plan 4.2, Appendix A, criteria related to seismic and LOCA loading conditions. The assembly structural integrity is assured.

CONDITION 3

An irradiation demonstration program should be performed to provide early confirmation performance data for the Vantage 5 design.

RESPONSE

A demonstration program was performed to determine early performance data for the Vantage 5 fuel assembly design features. The Vantage 5 demonstration program is described in Section 1 of Attachment 2 of Reference (1). Demonstration fuel rods of 15x15 design were irradiated in Turkey Points 3 and 4. Turkey Point 4 is currently operating with a full reload of fuel with many Vantage 5 features.

CONDITION 5

The WRB-2 correlation with a DNBR limit of 1.17 is acceptable for application to 17x17 Vantage 5 fuel. Additional data and analysis are required when applied to 14x14 or 15x15 fuel with an appropriate DNBR limit. The applicability range of WRB-2 is specified in Section 4.2.

RESPONSE

The WRB-2 correlation is not used since IP3 will use 15x15 Vantage 5 fuel without Intermediate Flow Mixer (IFM) grids. As described in Section 5.0 of Attachment (2) to Reference (1), the WRB-1 correlation with a DNBR limit of 1.17 is used for the Vantage 5 fuel with the Improved Thermal Design Procedure (ITDP) methodology.

ATTACHMENT III TO IPN-89-027

REVISED TECHNICAL SPECIFICATION PAGES
CONCERNING THE REACTOR COOLANT SYSTEM
FLOW RATE

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
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In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability with 95% confidence level that the minimum DNBR for the limiting rod is greater than or equal to the applicable DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. The DNBR uncertainty combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin is maintained by performing DNB design evaluations to a higher DNBR value, called the Safety Limit DNBR.

The curves of Figure 2.1-1 show the loci of points of thermal power, Reactor Coolant System pressure and vessel inlet temperature for which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The calculation of these limits includes:

1. $F^N_{\Delta H}$ of 1.56

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 332,240 gpm as measured at the plant,

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

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

$$F^N_{\Delta H} \leq 1.56\ [1 + 0.3\ (1-P)]$$

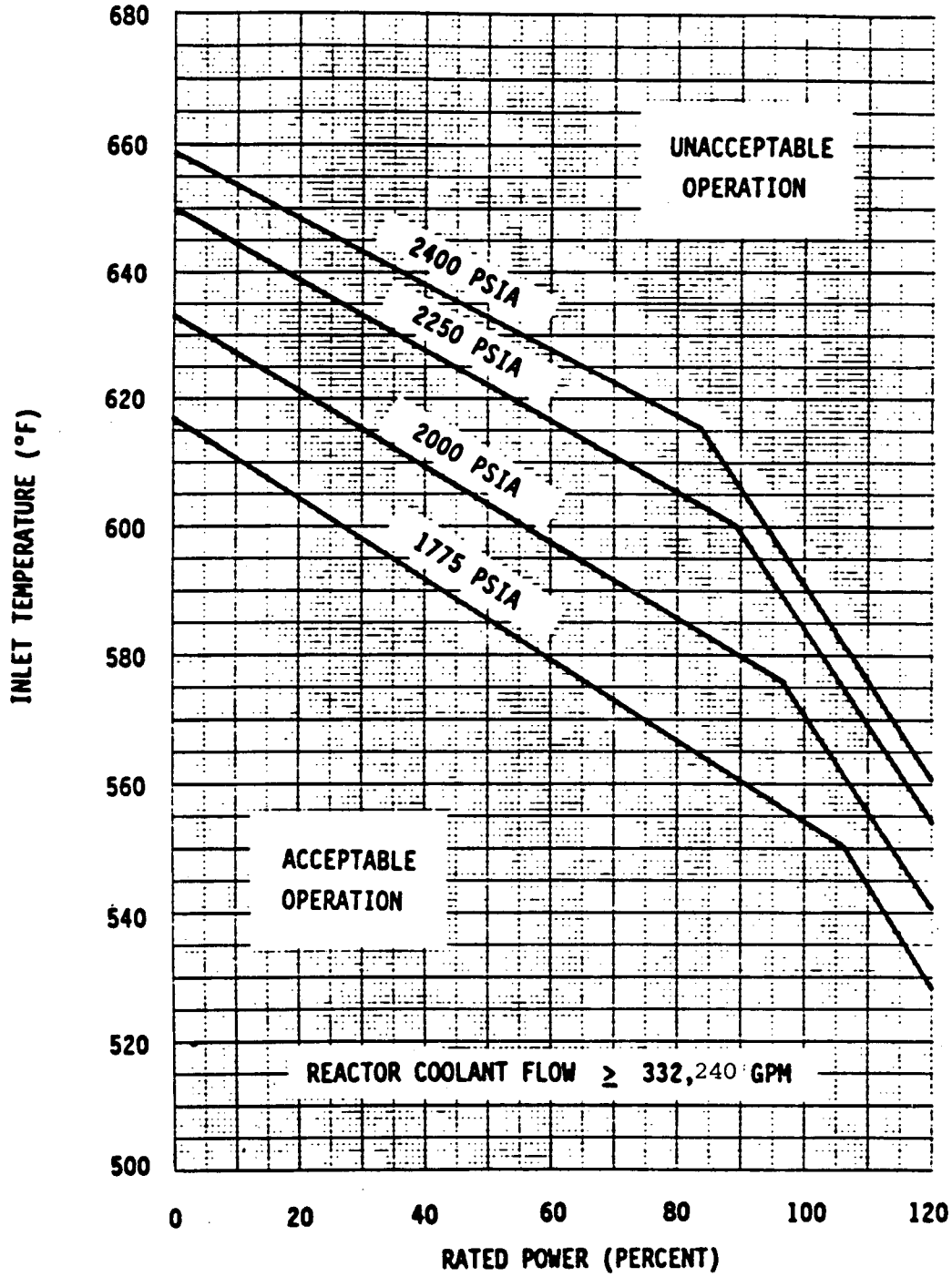
Where P is the fraction of Rated Thermal Power.

When flow or $F_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the limits presented. A 2.6% measurement uncertainty on Flow and a 4% measurement uncertainty of $F_{\Delta H}$ have already been included in the above limits.

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

References

1. FSAR Section 3.2.2.



100 PERCENT RATED POWER IS EQUIVALENT TO 3025 MWT

FIGURE 2.1-1