

In meeting the DNB criterion, uncertainties in operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB criterion.

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

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. $F_{\Delta H}^{RTP} = F_{\Delta H}^N$ limit at Rated Thermal Power (RTP) specified in the COLR.
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 375,600 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:

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

Where P is the fraction of Rated Thermal Power.

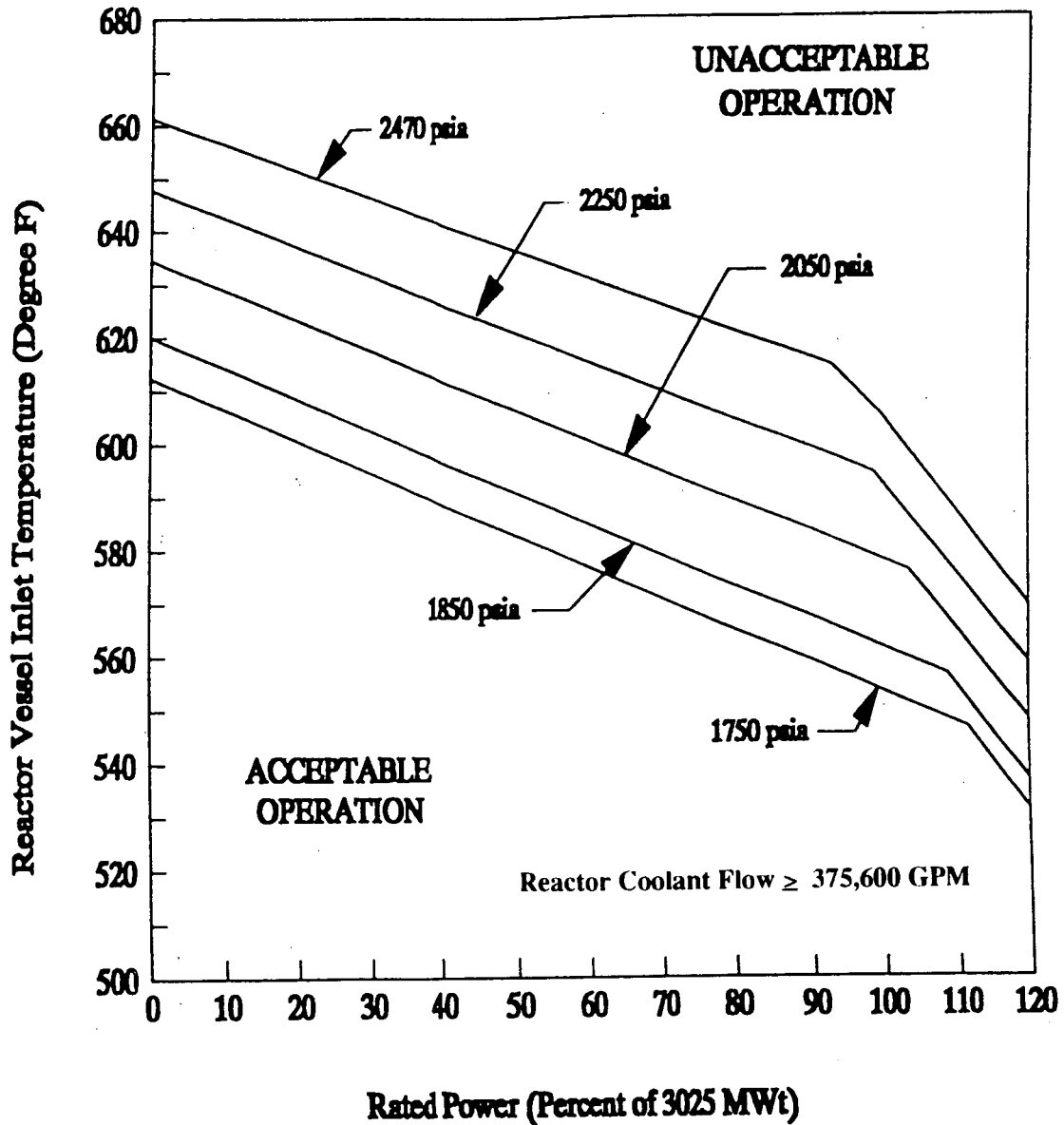
$F_{\Delta H}^{RTP}$ is the $F_{\Delta H}^N$ limit at Rated Thermal Power specified in the COLR, and $PF_{\Delta H}$ is the Power Factor Multiplier specified in the COLR.

When flow or $F_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the limits presented. A 2.9% measurement uncertainty on Flow and a 4% measurement uncertainty of $F_{\Delta H}$ have already been included in the above limits.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit (specified in the COLR) assuming the axial power imbalance is within the limits of the $f(\Delta I)$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

REACTOR CORE SAFETY LIMITS

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



100 PERCENT RATED POWER IS EQUIVALENT TO 3025 MWt

Pressures and temperatures do not include allowance for instrument error.

FIGURE 2.1-1

- ΔT_o \leq Measured full power ΔT for the channel being calibrated, °F
- T_{avg} = Average Temperature for the channel being calibrated, °F (input from instrument racks)
- T' = Measured full power T_{avg} for the channel being calibrated, °F
- P = Pressurizer pressure, psig (input from instrument racks)
- P' = 2235 psig (i.e., nominal pressurizer pressure at rated power)
- K_1 \leq 1.20
- K_2 = 0.0273
- K_3 = 0.0013
- K_1 is a constant which defines the overtemperature ΔT 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 ΔT setpoint to T_{avg} .
- K_3 is a constant which defines the dependence of the overtemperature ΔT setpoint to pressurizer pressure.
- ΔI = $q_t - q_b$, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power.
- $f(\Delta I)$ = a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are defined above such that:
- (a) for $q_t - q_b$ between -6.75 percent and +6.9 percent, $f(\Delta I) = 0$.
 - (b) for each percent that the magnitude of $q_t - q_b$ exceeds +6.9 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 3.333 percent of rated power.
 - (c) for each percent that the magnitude of $q_t - q_b$ is more negative than -6.75 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 4.000 percent of rated power.

3.1 Reactor Coolant System (RCS)

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

Specification

1. During the POWER OPERATION CONDITION, RCS DNB parameters for pressurizer pressure and RCS average temperature shall be within the limits specified below:
 - a. Pressurizer pressure \geq 2205 psig;
 - b. Maximum indicated $T_{avg} \leq$ 571.5°F; and
2. At the POWER OPERATION CONDITION with four reactor coolant pumps running, the RCS DNB parameter for RCS total flow rate shall be within the following limit:

RCS total flow rate \geq 375,600 gpm.
3. The pressurizer pressure limit of Specification 3.1.H.1 does not apply during:
 - a. THERMAL POWER ramp $>$ 5% RTP per minute; or
 - b. THERMAL POWER step $>$ 10% RTP.
4. If pressurizer pressure, RCS average temperature, or RCS total flow rate are not in accordance with Specifications 3.1.H.1, 3.1.H.2, or 3.1.H.3, then, immediately verify that the safety limits of Specification 2.1 have not been exceeded and, within 2 hours, restore the RCS DNB parameter(s) to within limits.
5. If pressurizer pressure and/or RCS average temperature are not restored to within limits within 2 hours, be in the HOT SHUTDOWN CONDITION within 6 hours.
6. If RCS total flow rate is not restored to within the limits of Specification 3.1.H.2 within 2 hours, bring THERMAL POWER to \leq 10% RTP within 6 hours and ensure operation is in accordance with Specification 3.1.A.1.e.

Surveillance Requirements

Reference Technical Specification Table 4.1-1, Items 4, 5, and 7, and Section 4.3.B.

Bases

Background

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS

3.1.H (continued)

pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure and temperature limits are consistent with operation within the nominal operational envelope. A lower pressure will cause the reactor core to approach DNB limits. A higher RCS average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit bounds that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

Applicable Safety Analyses

The requirements of this Specification represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this Specification will result in meeting the applicable DNBR criteria. Changes to the unit that could affect these parameters must be assessed for their effect on the DNBR criteria.

Specification

Specifications 3.1.H.1 and 3.1.H.2 specify limits on the monitored process variables (pressurizer pressure, RCS average temperature, and RCS total flow rate) to ensure that the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

The RCS total flow rate limit of 375,600 gpm allows for a measurement uncertainty of 2.9% associated with the performance of Reactor Coolant System Flow Calculation required by Technical Specification 4.3.B. Because the flow instrumentation provides flow indication based on a percentage of full flow, the 375,600 gpm is converted into a percentage of full flow to accommodate the verification that RCS total flow is within limits during channel checks.

The pressurizer pressure limit of 2205 psig allows for measurement uncertainty and instrument error. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit.

The limit on maximum indicated RCS average temperature provides assurance that RCS temperatures are maintained within the normal steady state envelope of operation assumed in the safety analyses performed to support the Vantage + fuel reloads with asymmetric tube

3.1.H (continued)

plugging among steam generators. A maximum full power T_{cold} of 547.7°F (including control deadband and measurement uncertainties) was assumed in these safety analyses. A T_{avg} of 578.3°F assures that a T_{cold} of 547.7°F is not exceeded at a measured flow of $\geq 375,600$ gpm when considering asymmetric tube plugging among steam generators for DNB considerations. However, T_{avg} will be controlled to a maximum indicated T_{avg} of 571.5°F which assures consistency with analyses for post-LOCA containment integrity.

Applicability

During the POWER OPERATION CONDITION, the limits on pressurizer pressure and RCS coolant average temperature must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. For the same reason, during the POWER OPERATION CONDITION with four reactor coolant pumps running, the limit on RCS flow rate must be maintained. In all other operating conditions, the power level is low enough that DNB is not a concern.

Specification 3.1.H.3 indicates that the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase $> 5\%$ RTP per minute or a THERMAL POWER step increase $> 10\%$ RTP. These conditions represent short term perturbations where actions to control pressure variations might be counter productive. Also, since they represent transients initiated from power levels $< 100\%$ RTP, an increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in Safety Limit 2.1, "Safety Limits, Reactor Core." Those limits are less restrictive than the limits of this specification but violation of a Safety Limit merits stricter, more severe required action. Should a violation of Specification 3.1.H.1 occur, the operator must check whether or not a Safety Limit has been exceeded.

Actions

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within specification limits, action must be taken to restore the parameter(s).

The 2 hour completion time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience for Westinghouse plants.

If the required action of Specification 3.1.H.4 is not met within the associated completion time, the plant must be brought to a mode in which Specification 3.1.H.1 does not apply. To achieve this status, the plant must be brought to at least the HOT SHUTDOWN CONDITION within 6 hours. The reduced power condition eliminates the potential for violation of the accident analysis bounds. The completion time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

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 infrequency over the life of the plant, to be associated primarily with determinations 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 calculations 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 hypothetical 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 assumed control rod drop time in the safety analysis is 2.7 seconds, consisting of 1.80 seconds for normal rod drop time plus additional margin which includes a seismic allowance. The required control rod drop time in Section 3.10.8 is therefore consistent with that assumed in the 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 23, 1985, entitled "Proposed Technical Specifications Regarding the Cycle 4/5 Refueling."

4.3 REACTOR COOLANT SYSTEM (RCS) TESTING

B. Reactor Coolant System Flow Calculation

Specification

Once every 24 months, prior to exceeding 24 hours of continuous operation with THERMAL POWER \geq 90% RTP, verify by flow calculation that RCS total flow rate is \geq 375,600 gpm.

Basis

Measurement of RCS total flow rate by performance of a flow calculation once every 24 months verifies that the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The frequency of 24 months reflects the importance of verifying flow after a refueling outage when the core has been altered or steam generator tubes have been plugged, which may have caused an alteration of flow resistance.

This specification allows for placement of the unit in the best condition for performing the Surveillance Requirement. The specification allows the Surveillance Requirement to be performed within 24 hours after THERMAL POWER \geq 90% RTP. This is appropriate because a flow calculation performed with the plant \geq 90% RTP will ensure that instrument inaccuracies are consistent with those assumed in the accident analyses. The Surveillance shall be performed within 24 hours of continuous operation at or above 90% RTP.

ATTACHMENT II TO IPN-97-023

**TECHNICAL JUSTIFICATION FOR PROPOSED TECHNICAL SPECIFICATION REVISIONS
ASSOCIATED WITH THE UPGRADE TO VANTAGE + FUEL**

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

**Technical Justification for Proposed Technical Specification Revisions
Associated With the Upgrade to VANTAGE + Fuel**

Measured Minimum Reactor Coolant System (RCS) Flow

The attached Technical Specification pages contain a revision to the minimum RCS flow from 385,400 gpm to 375,600 gpm. Subsequent to the submittal of Reference 1, fuel performance analyses were performed explicitly using a lower value (375,600 gpm) for measured minimum flow. This reduction provides additional operating margin, and still meets fuel performance limits.

Modification of $f(\Delta I)$ Function

The proposed change for the Overtemperature ΔT $f(\Delta I)$ function in the attached Technical Specification pages is consistent with the expected implementation of the relaxed axial offset control (RAOC) strategy for Cycle 10, as well as the increase in future cycle (Cycle 10 and beyond) lengths from 575 effective full power days (EFPD) to at least 630 EFPD. RAOC presents more limiting Condition I axial power shapes, relative to the constant axial offset control (CAOC) strategy (for which Indian Point 3 is currently licensed), for the initiation of Condition II transients, and hence results in more adverse Condition II transient shapes. The use of the $f(\Delta I)$ penalty with RAOC serves to limit the Condition II transients, and hence the axial power shapes, such that all departure from nucleate boiling (DNB), fuel centerline melt, and fuel rod design criteria are met. Since RAOC Condition I operation bounds CAOC, this $f(\Delta I)$ penalty is also directly applicable to CAOC operation.

Rod Drop Time Basis Change

The safety analysis value for rod drop time testing listed in the Technical Specification basis is being revised from 2.4 to 2.7 seconds. The applicable FSAR Chapter 14 (Reference 2) accident analyses which include rod drop time have been reevaluated using the new fuel features introduced by VANTAGE + fuel and the new safety analysis limit of 2.7 seconds. All performance criteria continue to be met.

The rod drop time of 1.8 seconds, contained in Technical Specification 3.10.8 (Reference 3), is not being revised. The current requirement of 1.8 seconds is conservative for transition cores of VANTAGE 5 and VANTAGE + fuel types.

References:

1. NYPA letter, W. J. Cahill, Jr. to the NRC (IPN-96-128), dated December 23, 1996, "Proposed Technical Specification Changes Associated With the Upgrade to VANTAGE + Fuel."
2. Indian Point 3 FSAR, Chapter 14.
3. Indian Point 3 Technical Specifications, Section 3.10.8.