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.

** A lower SG plugging level of 2% is presumed for the Loss of Normal Feedwater and Loss of AC Power analyses. A reduction in assumed SG tube plugging levels makes the curves in Figure 2.1-1 more conservative (i.e., adds greater margin).

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(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

2. "Safety Evaluation for Indian Point Unit 3 with Asymmetric Tube Plugging Among Steam Generators", WCAP-10705 (Westinghouse Non-Proprietary), October 1984.

ATTACHMENT II TO IPN-98-036

Evaluation of Change to the Technical Specification Basis Regarding Steam Generator Tube Plugging Limit

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

EVALUATION OF CHANGE TO THE TECHNICAL SPECIFICATION BASIS REGARDING STEAM GENERATOR TUBE PLUGGING LIMIT

The Auxiliary Feedwater (AFW) System is designed to supply auxiliary feedwater to the Steam Generators (SGs) for the removal of stored and residual heat from the Reactor Coolant System (RCS). Operation of the AFW system is required during normal operating conditions such as plant startup and cooldown, and accident conditions such as loss of normal feedwater events.

NYPA has recently discovered, through discussions with Westinghouse, that the purge volume associated with the AFW system is larger than previously assumed. The purge volume is the quantity of hot water which must be forced through the main feedwater lines before cooler water from the AFW system can reach the SGs. The purge volume used in previous analyses is equal to 190 ft^3 per main feedwater line. Recent calculations demonstrate that the actual purge volume is equal to 240.7 ft^3 per main feedwater line.

Incorporation of a larger purge volume into accident analyses increases the time interval between the time of reactor trip and the time at which cooler feedwater reaches the SGs. Therefore, a reanalysis of the affected accident scenarios (i.e., Loss of Normal Feedwater and Loss of AC Power to the Station Auxiliaries) was completed to ensure that the larger purge volume has no adverse effects on plant safety. These analyses were performed to demonstrate that the following applicable acceptance criteria are satisfied for these events:

- Reactor Coolant System (RCS) and Main Steam System (MSS) pressures shall be maintained below 110% of the applicable design pressures,
- the pressurizer shall not become water solid following the loss of heat sink accident, and
- the transient is analyzed for a sufficient duration to confirm that the pressurizer water level is decreasing, and a general cooling trend is observed. This demonstrates that the assumed AFW system conditions are capable of returning the plant to a safe condition by removing the stored and residual heat, thereby preventing either overpressurization or uncovery of the core.

If a 2% SG tube plugging limit is assumed, these analyses conclude that the AFW system has sufficient capacity to provide long term heat removal even assuming the limiting single active failure of a train of AFW actuation resulting in the availability of only one motor-driven pump supplying AFW to two-of-four steam generators. (Currently, the steam generator tube plugging levels are below 2%.) Therefore, the proposed TS basis changes state that a 2% SG tube plugging limit is assumed for the analysis of these two accident scenarios. The remainder of the accident analyses which are not affected by this change in the AFW purge volume are still based on the assumption of 24% uniform SG tube plugging with 30% asymmetric SG tube plugging.

The addition of this footnote is a temporary measure to ensure that the Technical Specifications accurately reflect the results of the recent accident analyses. As noted above, current plant steam generator tube plugging levels are below the 2% tube plugging limit specified in this basis change. However, the Authority is actively pursuing alternative options which will introduce additional margin into the analyses and ensure that the 2% tube plugging limit can be increased.

ATTACHMENT III TO IPN-98-036

Markup of Technical Specification Basis Pages

(FOR INFORMATION ONLY)

NOTE 1: Deletions are shown in ~~strikeout~~, and additions are shown in **bold**.

NOTE 2: Previous amendment revision bars are not shown.

NEW YORK POWER AUTHORITY
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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:

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

Where P is the fraction of Rated Thermal Power.

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