ENCLOSURE 3

NORTH ANNA UNITS 1 AND 2

TECHNICAL SPECIFICATION CHANGES

FOR

2905 MWt NSSS RATING

VIRGINIA POWER

JANUARY 1985

45-WBR-2010N-5

North Anna Core Uprating

(2893 MWt RATED THERMAL POWER)

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(1) Maximum Power Level

VEPCO is authorized to operate the North Anna Power Station, Unit No. 1, at reactor core power levels not in excess of 2893 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 63, are hereby incorporated in the license. VEPCO shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of this amendment or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission:

c. Virginia Electric and Power Company shall not operate the reactor in operational modes 1 and 2 with less than three reactor coolant pumps in operation.

e. If Virginia Electric and Power Company plans to remove or to make significant changes in the normal operation of equipment that controls the amount of radioactivity in effluents from the North Anna Station, the Commission shall be notified in writing regardless of whether the change affects the amount of radioactivity in the effluents.

QUADRANT POWER TILT RATIO

1.23 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2893 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.25 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.26 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.27 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.28 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of wet wastes into a solid form that meets shipping and burial ground requirements.

SOURCE CHECK

1.30 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

Nominal T_{avg} = 586.8°F

Nominal RCS flow = 289200 GPM

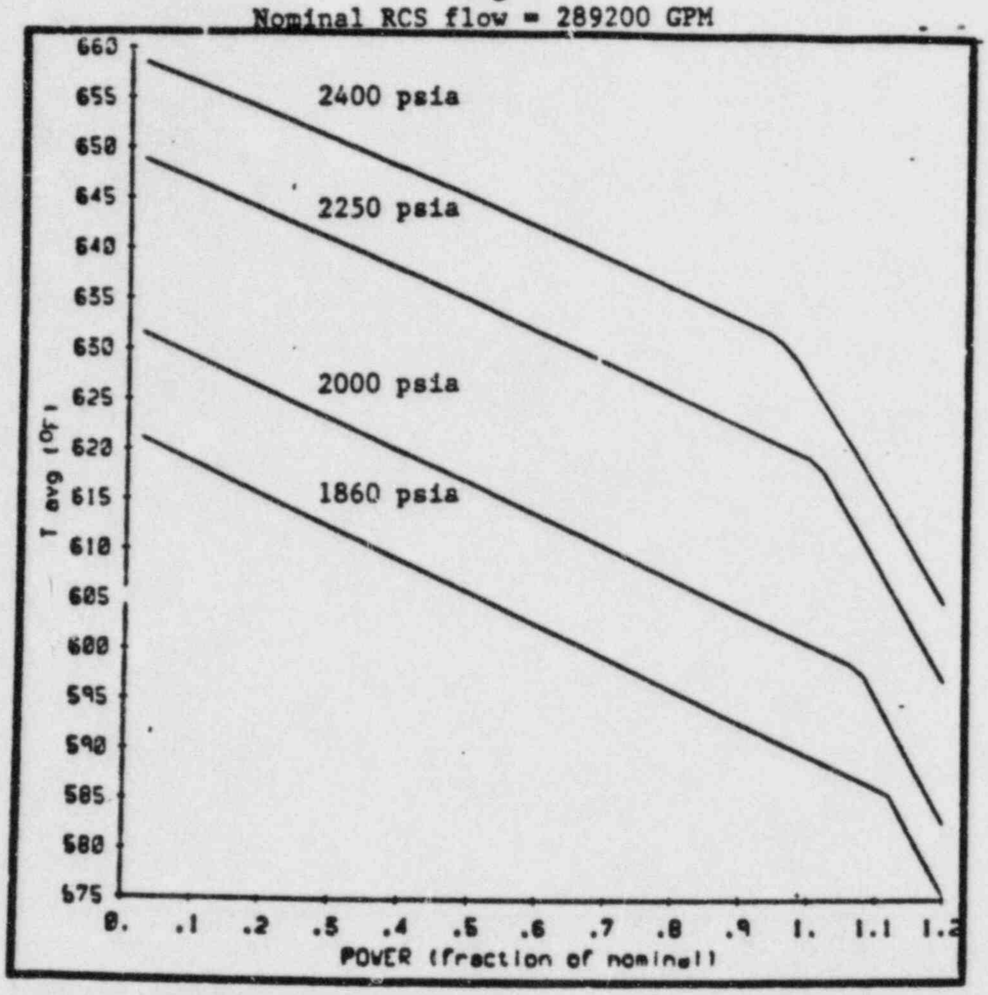

Figure 2.1-1 REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER
	High Setpoint- ≤ 109% of RATED THERMAL POWER	High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3×10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure-Low	≥ 1870 psig	≥ 1860 psig
10. Pressurizer Pressure-High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level-High	≤ 92% of instrument span	≤ 93% of instrument span
12. Loss of Flow	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

*Design flow is 96,400 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)(T-T') + K_3(P-P') - f_1(\Delta I)\right]$

Where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER ≤ 586.8°F

P = Pressurizer pressure, psig

P' = 2235 psig (indicated RCS nominal operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 25 secs, τ_2 = 4 secs.

S = Laplace transform operator (sec^{-1})

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated)*	Operation with 2 Loops (1 loop isolated)*
K_1 = 1.264	K_1 = ( )	K_1 = ( )
K_2 = 0.0220	K_2 = ( )	K_2 = ( )
K_3 = 0.001152	K_3 = ( )	K_3 = ( )

and f_1 (ΔI) 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 - 44 percent and + 3 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds - 44 percent, the ΔT trip setpoint shall be automatically reduced by 1.67 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + 3 percent, the ΔT trip setpoint shall be automatically reduced by 2.00 percent of its value at RATED THERMAL POWER.

*Values dependent on NRC approval of ECCS evaluation for these operating conditions.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 2: Overpower $\Delta T \leq \Delta T_o [K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T-T') - f_2(\Delta I)]$

Where:

ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER ≤ 586.8°F

K_4 = 1.079

K_5 = 0.02/°F for increasing average temperature

K_5 = 0 for decreasing average temperatures

K_6 = 0.00164 for $T > T'$; K_6 = 0 for $T \leq T''$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator (sec^{-1})

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent span.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

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 correlation. The W-3 DNB correlation 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 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 (the WRB-1 correlation in this application). 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.

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 that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This 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.

The curves of Figures 2.1-1, 2.1-2 and 2.1-3 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the design limit DNBR, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$, at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49\ [1+0.3\ (1-P)]$$

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 assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature 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.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, were initially designed to ANSI B 31.1 1967 Edition and ANSI B 31.7 1969 Edition (Table 5.2.1-1 of FSAR) which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

LIMITING SAFETY SYSTEM SETTINGS

BASES

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBR's will be greater than the applicable design limit DNBR value for each fuel type.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System. The pressurizer high water level trip is blocked automatically below the P-7 setpoint.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90% of nominal full loop flow. Above 31% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the design limit during normal operational transients and anticipated transients when 2 loops are in operation and the Overtemperature ΔT trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT trip setpoint adjusted to the value specified for 2 loop operation, the P-8 trip at 71% RATED THERMAL POWER with the loop stop valves closed in the nonoperating loop, will prevent the minimum value of the DNBR from going below the design limit during normal operational transients with 2 loops in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system. The steam generator water level low-low trip is blocked when the loop stop valves are closed. A steam generator water level high-high signal trips the turbine which in turn trips the reactor if above the P-7 setpoint.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.15}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.30] \; [K(Z)] \text{ for } P \leq 0.5$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Comply with either of the following ACTIONS:

1. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

2. Reduce THERMAL POWER as necessary to meet the limits of Specification 3.2.6 using the APDMS with the latest incore map and updated $\overline{R}$.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

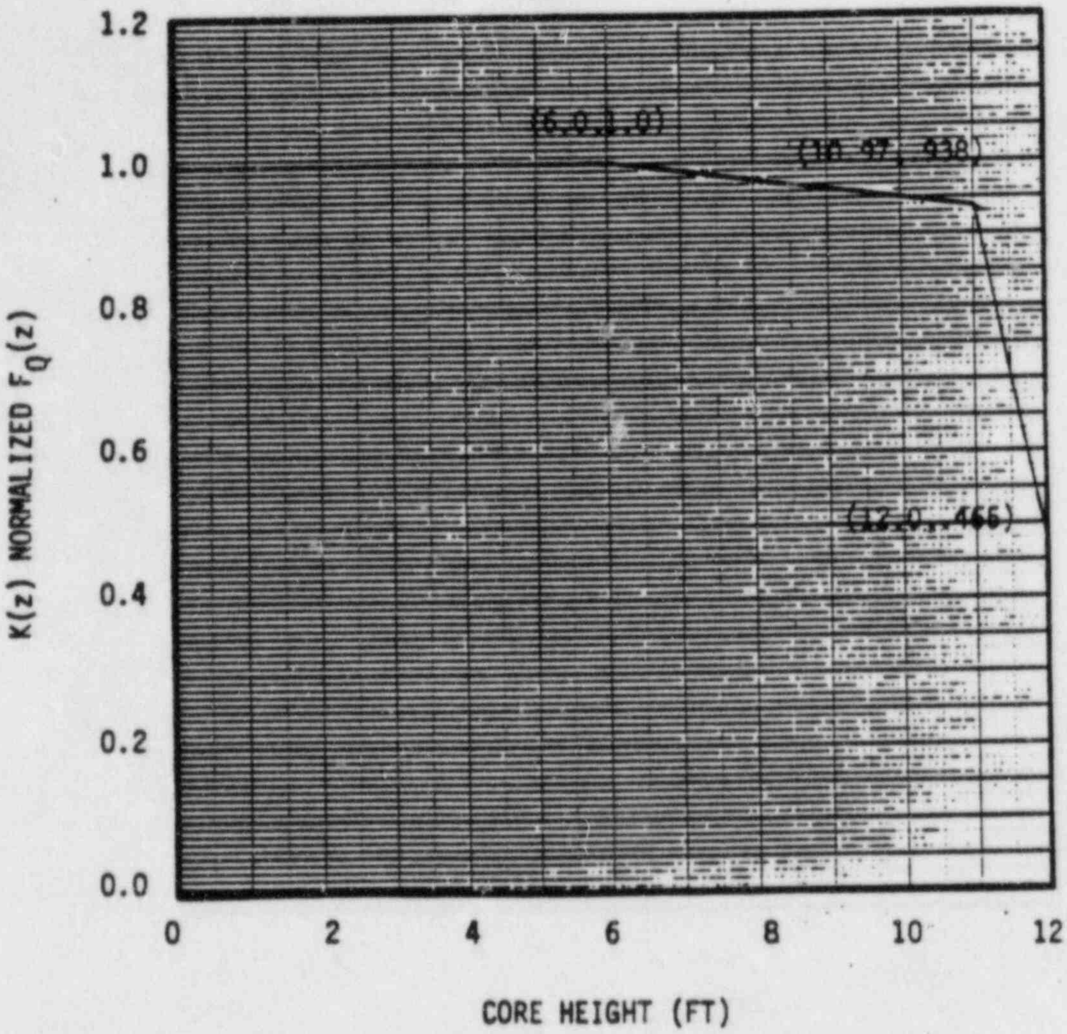

Figure 3.2-2 NORMALIZED $F_Q(z)$ AS A FUNCTION OF CORE HEIGHT

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.49 \; [1 + 0.3 \; (1-P)]$$

$$\text{where: } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

$F^N_{\Delta H}$ = measured value of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map.

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The provisions of Specification 4.0.4 are not applicable.

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TABLE 3.2-1

DNB PARAMETERS

PARAMETER	3 Loops in Operation	LIMITS 2 Loops in Operation** & Loop Stop Valves Open	2 Loops in Operation** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	≤591°F		
Pressurizer Pressure	≥2205 psig*		
Reactor Coolant System Total Flow Rate	≥289,200 gpm		

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

** Values dependent on NRC approval of ECCS evaluation for these conditions.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core from going beyond the design limit DNBR during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ 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.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope, as given in Specification 3.2.2, is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS - $F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2. and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than ±12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

The specified limit for $F^N_{\Delta H}$ contains a 4% error allowance. Normal operation will result in a measured $F^N_{\Delta H} \leq 1.49$. The 4% allowance is based on the following considerations:

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the margin available between the safety analysis design DNBR values (1.57 and 1.59 for thimble and typical cells, respectively) and the limiting design DNBR values (1.39 for thimble cells and 1.42 for typical cells). The applicable value of rod bow penalties can be obtained from the FSAR.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbles. The two sets of 4 symmetric thimbles is a unique set of 8 detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than the design limit throughout each analyzed transient. Measurement uncertainties must be accounted for during the periodic surveillance.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.2.6 AXIAL POWER DISTRIBUTION

The limit on axial power distribution ensures that F_Q will be controlled and monitored on a more exact basis through use of the APDMS when operating above P_m% of RATED THERMAL POWER. This additional limitation on F_Q is necessary in order to provide assurance that peak clad temperatures will remain below the ECCS acceptance criteria limit of 2200°F in the event of a LOCA. The value for P_m is based on the cycle dependent potential violation of the F_QxK(Z) limit, where K(Z) is the graph shown in Figure 3.2-2. The amount of potential violation is determined by subtracting 1 from the maximum ratio of the predicted F_Q(Z) analysis (flyspeck) results for a particular fuel cycle to the F_QxK(Z) limit. This amount of potential violation, in percent, is subtracted from 100% to determine the value for P_m. If P_m is equal to 100%, no axial power distribution surveillance is required. P_m will not exceed 100%.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensure that the secondary system pressure will be limited to within 110% of the system design pressure, during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all safety valves on all of the steam lines is 12.83×10^6 lbs/hr which is greater than the total secondary steam flow of 12.77×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 3 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 109$$

For 2 loop operation with stop valves closed

$$SP = \frac{(X) - (Y)\ (U)}{X} \times 71$$

For 2 loop operations with stop valves open

$$SP = \frac{(X) - (Y)\ (U)}{X} \times 66$$

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, VEPCO to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, VEPCO to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Vepco is authorized to operate the facility at steady state reactor core power levels not in excess of 2893 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 47, are hereby incorporated in the license. VEPCO shall operate the facility in accordance with the Technical Specifications.

(b) The current surveillance period for Surveillance Requirement 4.7.10.c may be extended beyond the time limit specified by Technical Specification 4.0.2.a. The required surveillance shall be completed prior to startup after the first refueling outage. The plant shall not be operated in Modes 1, 2, 3 or 4 until Surveillance Requirement 4.7.10.c has been completed. Upon accomplishment of the surveillance, the provisions of 4.0.2.a shall apply.

QUADRANT POWER TILT RATIO

1.23 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2893 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.25 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.26 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.27 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.28 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of wet wastes into a solid form that meets shipping and burial ground requirements.

SOURCE CHECK

1.30 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

Nominal T_{avg} = 586.8°F

Nominal RCS flow = 289200 GPM

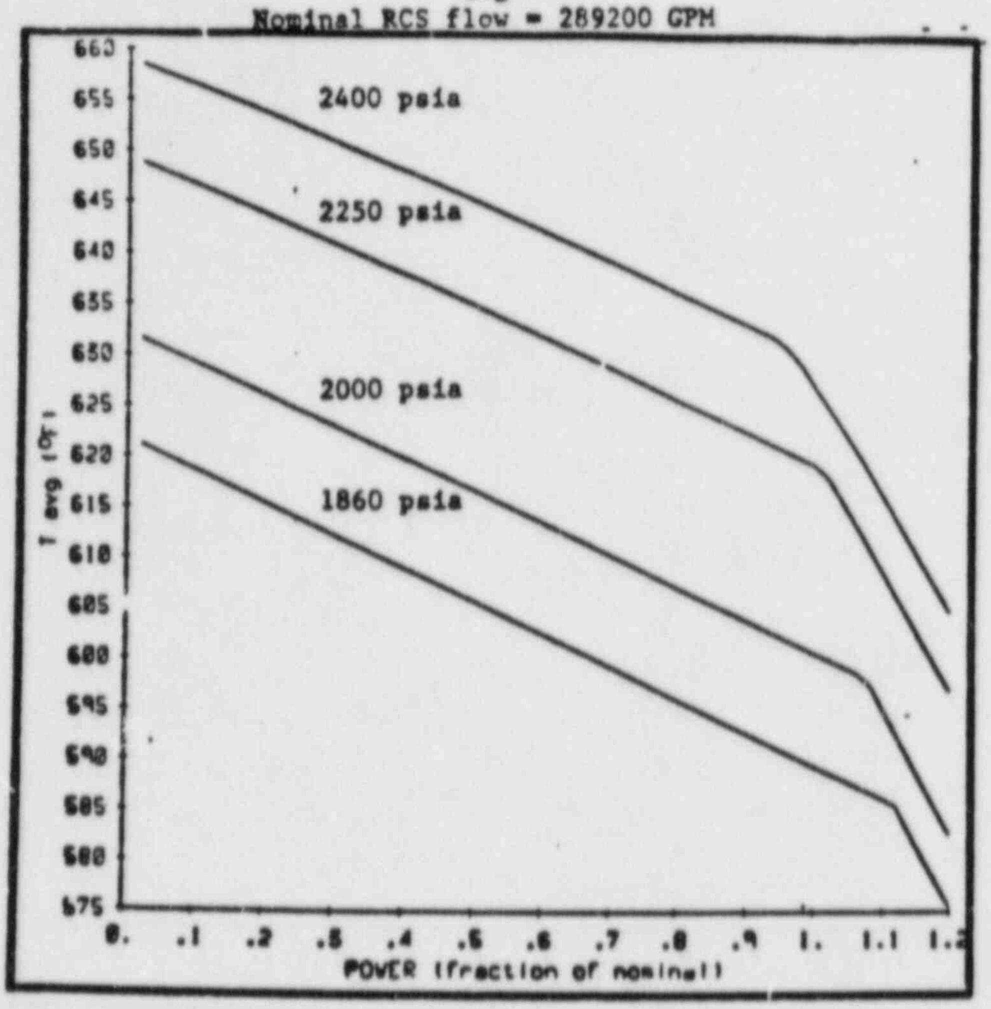

Figure 2.1-1 **REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION**

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER
	High Setpoint- ≤ 109% of RATED THERMAL POWER	High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3×10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure-Low	≥ 1870 psig	≥ 1860 psig
10. Pressurizer Pressure-High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level-High	≤ 92% of instrument span	≤ 93% of instrument span
12. Loss of Flow	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

*Design flow is 96,400 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right) (T-T') + K_3(P-P') - f_1(\Delta I)\right]$

Where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER ≤ 586.8°F

P = Pressurizer pressure, psig

P' = 2235 psig (indicated RCS nominal operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

$\tau_1 + \tau_2$ = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 25 secs, τ_2 = 4 secs.

S = Laplace transform operator (sec^{-1})

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated)*	Operation with 2 Loops (1 loop isolated)*
K_1 = 1.264	K_1 = ( )	K_1 = ( )
K_2 = 0.0220	K_2 = ( )	K_2 = ( )
K_3 = 0.001152	K_3 = ( )	K_3 = ( )

and f_1 (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 - 44 percent and + 3 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of (q_t - q_b) exceeds - 44 percent, the ΔT trip setpoint shall be automatically reduced by 1.67 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of (q_t - q_b) exceeds + 3 percent, the ΔT trip setpoint shall be automatically reduced by 2.00 percent of its value at RATED THERMAL POWER.

*Values dependent on NRC approval of ECCS evaluation for these operating conditions.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 2: Overpower $\Delta T \leq \Delta T_o [K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T-T'') - f_2(\Delta I)]$

Where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T'' = Indicated T_{avg} at RATED THERMAL POWER ≤586.8°F

K_4 = 1.079

K_5 = 0.02/°F for increasing average temperature

K_5 = 0 for decreasing average temperatures

K_6 = 0.00164 for $T > T''$; K_6 = 0 for $T \leq T''$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator (sec^{-1})

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent span.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

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 building (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 correlation. The W-3 DNB correlation 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 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 (the WRB-1 correlation in this application). 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.

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 that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This 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.

The curves of Figures 2.1-1, 2.1-2 and 2.1-3 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the design limit DNBR, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49\ [1+0.3\ (1-P)]$$

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 assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature 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.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, were initially designed to ANSI B 31.1 1967 Edition and ANSI B 31.7 1969 Edition (Table 5.2.1-1 of FSAR) which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBR's will be greater than the applicable design limit DNBR value for each fuel type.

This latter trip will prevent the minimum value of the DNBR from going below the design limit during normal operational transients and anticipated transients when 2 loops are in operation and the Overtemperature ΔT trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT trip setpoint adjusted to the value specified for 2 loop operation, the P-8 trip at 71% RATED THERMAL POWER with the loop stop valves closed in the nonoperating loop, will prevent the minimum value of the DNBR from going below the design limit during normal operational transients with 2 loops in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system. The steam generator water level low-low trip is blocked when the loop stop valves are closed. A steam generator water level high-high signal trips the turbine which in turn trips the reactor if above the P-7 setpoint.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than 1.616×10^6 lbs/hour of full steam flow at RATED THERMAL POWER. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 25 percent, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.15}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.30] \quad [K(Z)] \text{ for } P \leq 0.5$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Comply with either of the following ACTIONS:

1. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

2. Reduce THERMAL POWER as necessary to meet the limits of Specification 3.2.6 using the APDMS with the latest incore map and updated $\bar{R}$.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

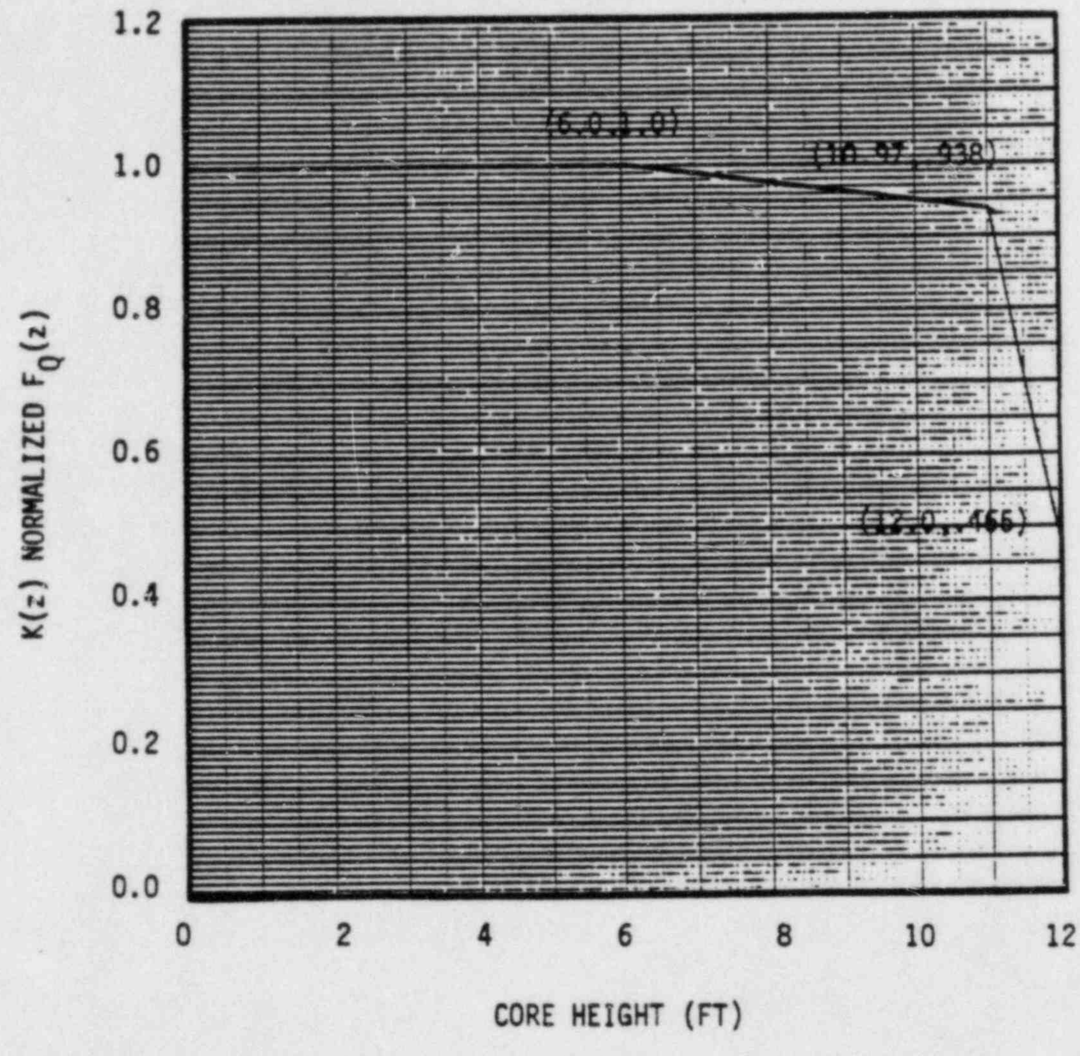

Figure 3.2-2 NORMALIZED $F_Q(z)$ AS A FUNCTION OF CORE HEIGHT

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.49 \; [1 + 0.3 \; (1-P)]$$

$$\text{where: } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

$F^N_{\Delta H}$ = measured value of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map.

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to

ACTION Continued

exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The provisions of Specification 4.0.4 are not applicable.

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TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS		
	3 Loops in Operation	2 Loops in Operation** & Loop Stop Valves Open	2 Loops in Operation** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq$591°F		
Pressurizer Pressure	$\geq$2205 psig*		
Reactor Coolant System Total Flow Rate	$\geq$289,200 gpm		

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

** Values dependent on NRC approval of ECCS evaluation for these conditions.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core from going beyond the design limit DNBR during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ 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.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope, as given in Specification 3.2.2, is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other

The specified limit for $F^N_{\Delta H}$ contains a 4% error allowance. Normal operation will result in a measured $F^N_{\Delta H}$ less than or equal to 1.49. The 4% allowance is based on the following considerations:

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distribtions. This compensation for $F^N_{\Delta H}$ is less readily available.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the margin available between the safety analysis design DNBR values (1.57 and 1.59 for thimble and typical cells, respectively) and the limiting design DNBR values (1.39 for thimble cells and 1.42 for typical cells). The applicable value of rod bow penalties can be obtained from the FSAR.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbles. The two sets of 4 symmetric thimbles is a unique set of 8 detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than the design limit throughout each analyzed transient. Measurement uncertainties must be accounted for during the periodic surveillance.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.2.6 AXIAL POWER DISTRIBUTION

The limit on axial power distribution ensures that F_Q will be controlled and monitored on a more exact basis through use of the APDMS when operating above P_m% of RATED THERMAL POWER. This additional limitation on F_Q is necessary in order to provide assurance that peak clad temperatures will remain below the ECCS acceptance criteria limit of 2200^{o}F in the event of a LOCA. The value for P_m is based on the cycle dependent potential violation of the F_QxK(Z) limit, where K(Z) is the graph shown in Figure 3.2-2. The amount of potential violation is determined by subtracting 1 from the maximum ratio of the predicted F_Q(Z) analysis (flyspeck) results for a particular fuel cycle to the F_QxK(Z) limit. This amount of potential violation, in percent, is subtracted from 100% to determine the value for P_m. If P_m is equal to 100%, no axial power distribution surveillance is required. P_m will not exceed 100%.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensure that the secondary system pressure will be limited to within 110% of the system design pressure, during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all safety valves on all of the steam lines is 12.83×10^6 lbs/hr which is greater than the total secondary steam flow of 12.77×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 3 loop operation

$$SP = \frac{(X) \quad (Y)(V)}{X} \times 109$$

For 2 loop operation with stop valves closed

$$SP = \frac{(X) \quad (Y)\ (U)}{X} \times 71$$

ENCLOSURE 4

NORTH ANNA UNITS 1 AND 2

2905 MWt NSSS RATING

RESPONSE TO NRC COMMENTS
RAISED AT THE OCTOBER 11, 1984 MEETING

VIRGINIA POWER

JANUARY 1985

45-WBR-2010N-6

RESPONSE TO NRC STAFF COMMENTS RAISED AT THE OCTOBER 11, 1984 MEETING CORE UPRATING TO 2905 MWt NORTH ANNA POWER STATION UNITS 1 AND 2

I. Comment: Will Virginia Power update the Environmental Report as part of the request for an amendment to the Operating Licenses?

Response: The Environmental Report for North Anna Power Station (Appendix K) and the NRC's Environmental Impact Statement (Reference 1) have already addressed operation up to the stretch rating of 2910 MWt. Therefore, Virginia Power does not deem it necessary to revise the Environmental Report. It is recognized, however, that the temperatures in Lake Anna have exceeded predictions originally made in the Environmental Report and that the lake, at times, has exceeded State thermal limits. Virginia Power is currently conducting a 316(a) demonstration to resolve whether or not these temperatures adversely affect aquatic life. This study is being conducted in accordance to Section 316(a) of the Clean Water Act. The proposed core uprating is projected to increase the rejected heat by approximately 4.5%. The Virginia State Water Control Board and their Technical Advisory Committee are aware of the proposed core uprating and will consider it as they consider the results of the 316(a) demonstration.

II. Comment: Will the licensing submittal address the NRC's generic concerns regarding consistency between the Station Technical Specifications and accident analyses?

Response: Virginia Power has reviewed NRC concerns regarding Technical Specifications for: 1) Mode 3, 4 and 5 operation; 2) safety related equipment; 3) comparison of indicated parameter values with limits and 4) use of Non-Docketed documents to establish assumed initial conditions for analyses. These concerns have been reviewed for potential impact upon the proposed Technical Specification changes for uprated operation. It has been concluded that these concerns do not require any revision in the proposed changes for uprated operation. However, an additional submittal may be required to address the outstanding item discussed below. The Westinghouse Owners' Group (WOG) is considering the appropriate generic approach which should be taken to address these concerns for existing plant Technical Specifications. Virginia Power has therefore only addressed the generic issues as they potentially impact this licensing submittal, so as not to propose any resolution of these items which may be inconsistent with that determined through the WOG efforts.

The uprating analysis of the uncontrolled rod bank withdrawal from a subcritical condition was performed assuming two reactor coolant pumps are in operation. Virginia Power has separately prepared a submittal (Reference 4) to support operation with a positive moderator temperature coefficient. The subcritical rod withdrawal results in this separate submittal bound operation at current plant conditions with a single coolant pump operating. The final Westinghouse analysis methodology for this case was not available for the uprating submittal. Furthermore, Westinghouse informed the NRC via Reference 2 that they do not plan to pursue development of a generic methodology which assumes only one reactor coolant pump in operation. A future submittal is proposed to provide results of a Virginia Power analysis of this event at uprated conditions.

III. Comment: The NRC has stated that the core uprating request is likely to be treated as involving a significant hazards consideration.

Response: The evaluations performed to support uprated operation have addressed all items which were identified to be impacted by the proposed power increase. In addition, the proposed power level was chosen to remain within existing margins for plant analyses and equipment operation. The results of the evaluations performed confirm that all accidents, systems and equipment remain within the appropriate acceptance criteria. On this basis, the small increase in power is not considered to represent a significant hazard.

IV. Comment: Relevant to the core uprating, how is Virginia Power planning to address the NRC staff concern on the environmental qualification of equipment in areas that are potentially exposed to superheated steam? Note that if the steam generator tubes are uncovered during a steam line break, superheated steam could be released and this may not have been considered in past analyses.

Response: In response to NRC questions during review of Reference 3, Westinghouse has addressed the effects of steam generator tube uncovery on containment temperature for steam line breaks inside containment. It was concluded that existing analyses remain bounding for steam line breaks inside dry containments. A Virginia Power review of confined areas outside of containment was conducted to determine those areas potentially subject to steam releases from a steam line break. It was concluded that the Main Steam Valve House is the only such area containing equipment requiring environmental qualification. The existing MSLB calculation for this zone does not specifically address the existence of superheated steam, but the Virginia Power evaluation

concluded that the existing temperature envelope bounds the expected results in the presence of superheated steam. These results are being incorporated into the Environmental Zone Description updates currently being performed by Virginia Power. Note that, as stated in the BOP Safety Evaluation, the core uprate will not impact steam line break environments inside or outside the containment since those breaks are based on the limiting no-load power conditions which remain unaffected by the uprate.

V. Comment: The licensing submittal should contain enough information so the NRC staff can determine that the Station Design Basis criteria will continue to be met for the core uprating.

Response: The NSSS and BOP Licensing summaries document the evaluations performed in sufficient detail to demonstrate that the Station Design Basis Criteria will not be violated as a result of the uprating. Note, however, that the use of the Westinghouse Improved Thermal Design Procedure (ITDP) for DNB analysis represents a change in the design basis for DNB. The reanalysis of DNB-limited events has confirmed that the revised design basis is met. Appendix R evaluations are being performed for North Anna Power Station Units 1 and 2. These evaluations are not addressed as part of this core uprating submittal; however, the Appendix R evaluations will support operation up to a NSSS Rating of 2905 MWt. The Appendix R Report (Volume I), submitted to the NRC in Reference 5, lists the Core Uprating as one of the assumptions for the Appendix R Analyses.

VI. References:

1. Final Environmental Impact Statement, North Anna Power Station, U.S. Atomic Energy Commission, Directorate of Licensing, April 1973

2. Letter from E. P. Rahe (Westinghouse) to H. Thompson (NRC), NS-NRC-85-2997, dated January 22, 1985, Subject: Number of Operating Reactor Coolant Pumps in Mode 3

3. WCAP-8822, "Mass and Energy Releases Following Steam Line Rupture"

4. Letter from W. L. Stewart (Virginia Power) to H. R. Denton (NRC), dated February 7, 1985 (Serial No. 666), Subject: Positive Moderator Temperature Coefficient, Amendment to Operating Licenses NPF-4 and NPF-7, North Anna Power Station, Units 1 and 2

5. Letter from W. L. Stewart (Virginia Power) to H. R. Denton (NRC), dated March 8, 1985 (Serial No. 85-114), Subject: 10 CFR 50 Appendix R Reanalysis, North Anna Power Station Unit Nos. 1 and 2