FIGURE 1

SURRY UNIT 1 CYCLE 7
CONSERVATIVE CALCULATION OF ENTHALPY RISE FACTOR*
WITH POWER LEVEL AND
TECHNICAL SPECIFICATION LIMITS

*BASED ON RAISED INSERTION LIMITS

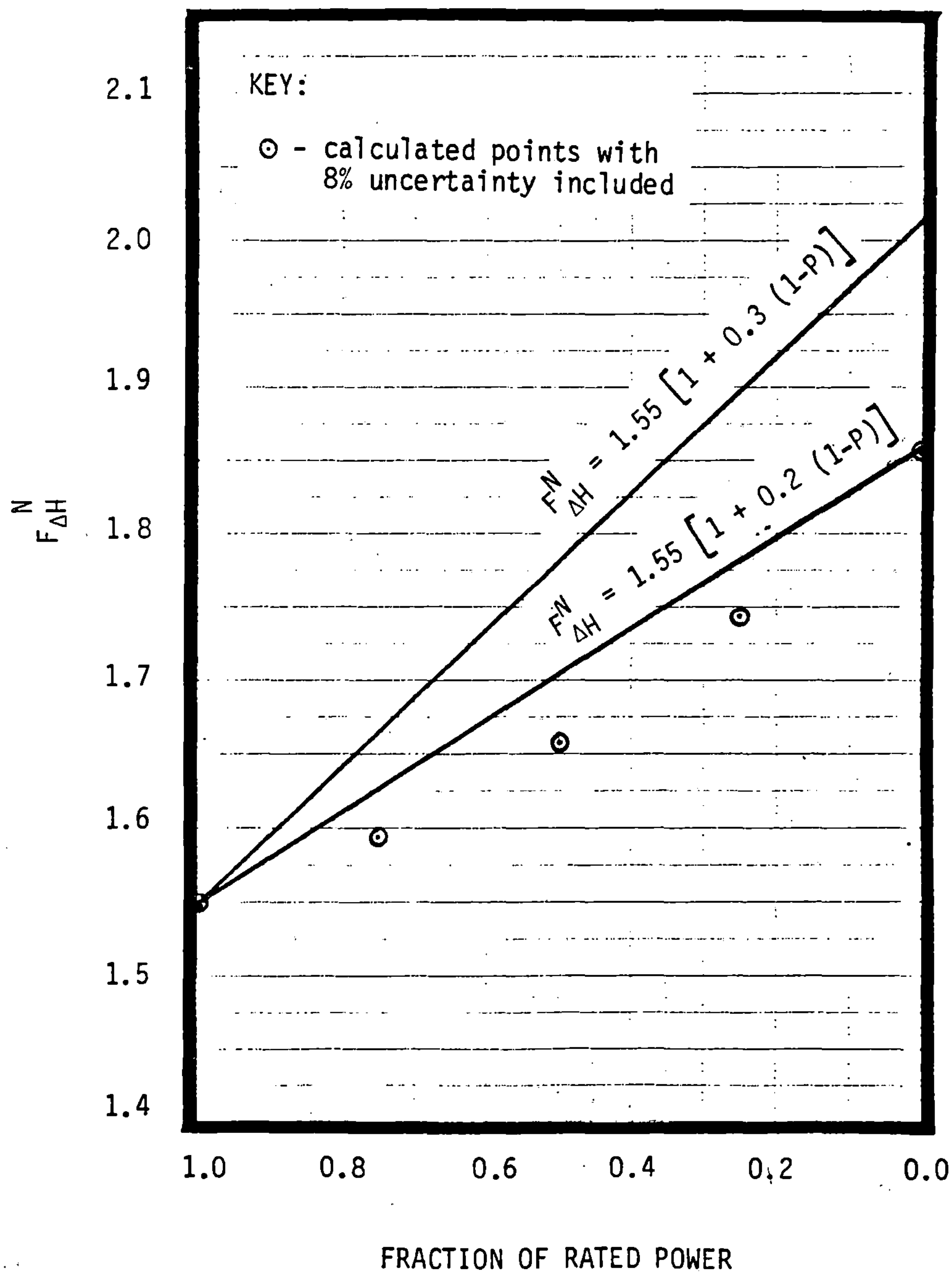
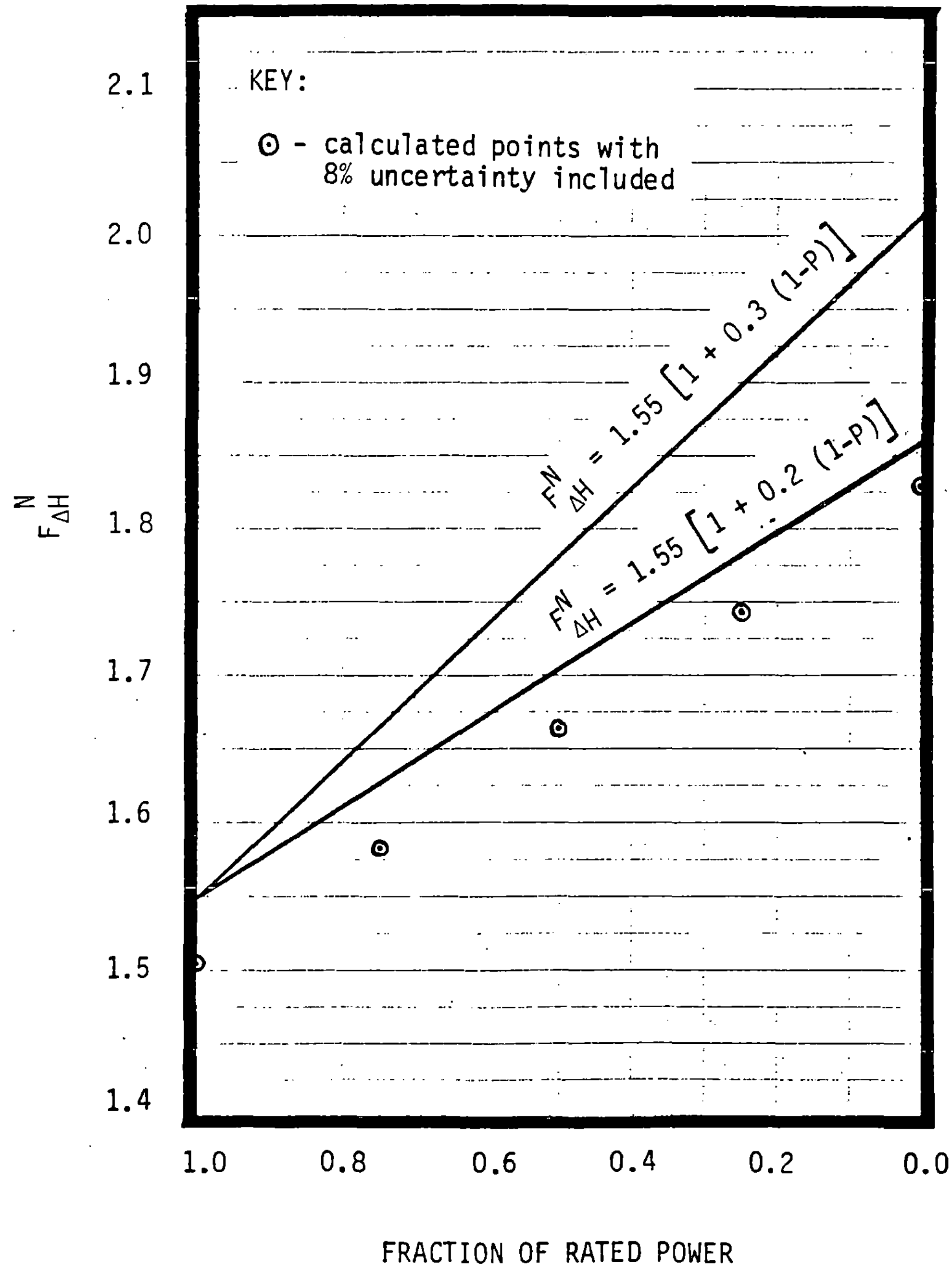


FIGURE 2

SURRY UNIT 2 CYCLE 7
CONSERVATIVE CALCULATION OF ENTHALPY RISE FACTOR
WITH POWER LEVEL AND
TECHNICAL SPECIFICATION LIMITS



ENCLOSURE 2

SURRY TECHNICAL SPECIFICATION CHANGES

uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNB ratio (DNBR) during steady state operation, normal operational transients and anticipated transients, is limited to 1.30. A DNBR of 1.30 corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.(1)

The curves of TS Figure 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (three loop operation) represent limits equal to, or more conservative than, the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which the DNB ratio is equal to 1.30 or the average enthalpy at the exit of the vessel is equal to the saturation value. The area where clad integrity is assured is below these lines. The temperature limits are considerably more conservative than would be required if they were based upon a minimum DNB ratio of 1.30 alone but are such that the plant conditions required to violate the limits are precluded by the self-actuated safety valves on the steam generators. The three loop operation safety limit curve has been revised to allow for heat flux peaking effects due to fuel densification and to apply to 100% of design flow. The effects of rod bowing are also considered in the DNBR analyses.

The curves of TS Figures 2.1-2 and 2.1-3 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation), represent limits equal to, or more conservative, than the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNB ratio is equal to 1.30 or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNB ratio reaches 1.30 and, thus, this arbitrary limit is conservative with respect to maintaining clad integrity. The plant conditions required to violate these limits are precluded by the protection system and the self-actuated safety valves on the steam generator. Upper limits of 70% power for loop stop valves open and 75% with loop stop valves closed are shown to completely bound the area where clad integrity is assured. These latter limits are arbitrary but cannot be reached due to the Permissive 8 protection system setpoint which will trip the reactor on high nuclear flux when only two reactor coolant pumps are in service.

Operation with natural circulation or with only one loop in service is not allowed since the plant is not designed for continuous operation with less than two loops in service.

TS Figures 2.1-1 through 2.1-3 are based on a $F^N_{\Delta H}$ of 1.55, a 1.55 cosine axial flux shape and a DNB analysis procedure including the fuel densification power spiking (4) as part of the generic margin to accommodate rod bowing (5)(6). TS Figure 2.1-1 is also valid for the following limit of the enthalpy rise hot channel factor: $F^N_{\Delta H} = 1.55\ (1 + 0.3\ (1-P))$ where P is the fraction of rated power. TS Figures 2.1-2 and 2.1-3 include a 0.2 rather than 0.3 part power multiplier for the enthalpy rise hot channel factor.

These hot channel factors are higher than those calculated at full power over the range between that of all control rod assemblies fully withdrawn to to this limiting criterion. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of this limit.

References

1) FSAR Section 3.4

2) FSAR Section 3.3.

3) FSAR Section 14.2

4) WCAP-8012, "Fuel Densification-Surry Power Station", December 1972 Section 4.3

5) Westinghouse (C. Eicheldinger) to NRC (V. Stello) letter dated August 13, 1976, Serial No. NS-CE-1163

6) NRC (A. Schwencer) to Vepco (W. L. Proffitt) letter dated July 27, 1979

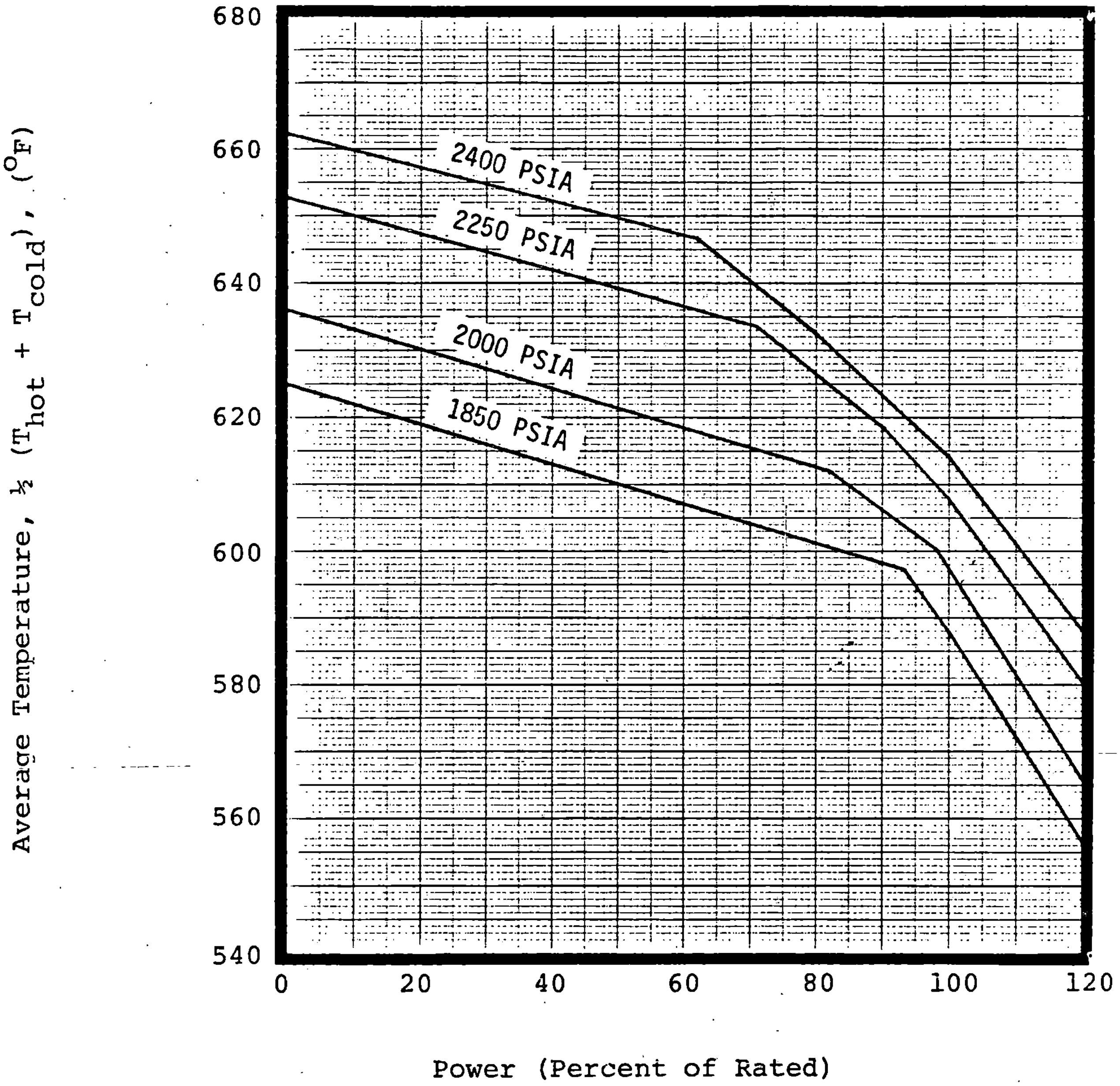


FIGURE 2.1-1 Reactor Core Thermal & Hydraulic Safety Limits
Three Loop Operation, 100% Flow

(b) High pressurizer pressure - $\leq$ 2385 psig.
(c) Low pressurizer pressure - $\geq$ 1860 psig.
(d) Overtemperature ΔT

$$\Delta T \leq \Delta T_o \left[K_1 - K_2\left(\frac{1 + t_1 S}{1 + t_2 S}\right)(T - T') + K_3(P - P') - f(\Delta I)\right]$$

where

T_o = Indicated T at rated thermal power, °F

T = Average coolant temperature, °F

T'= 574.4°F

P = Pressurizer pressure, psig

P' = 2235 psig

K_1 = 1.135
K_2 = 0.01072
K_3 = 0.000566 for 3-loop operation

K_1 = 0.951
K_2 = 0.01012 for 2-loop operation with loop stop
K_3 = 0.000554 valves open in inoperable loop

K_1 = 1.026
K_2 = 0.01012 for 2-loop operation with loop stop
K_3 = 0.000554 valves closed in inoperable loop

$\Delta I = q_t - q_b$, 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

$f(\Delta I)$ = function of ΔI, percent of rated core power as shown in Figure 2.3-1

t_1 = 25 seconds

t_2 = 3 seconds

(e) Overpower ΔT

$$\Delta T \leq \Delta T_o \left[K_4 - K_5\left(\frac{t_3 S}{1 + t_3 S}\right) T - K_6 (T - T') - f(\Delta I)\right]$$

where

ΔT_o = Indicated ΔT at rated thermal power, °F

T = Average coolant temperature, °F

T' = Average coolant temperature measured at nominal conditions and rated power, °F

K_4 = A constant = 1.089

K_5 = 0 for decreasing average temperature

A constant, for increasing average temperature 0.02/°F

K_6 = 0 for $T \leq T'$

= 0.001086 for $T > T'$

$f(\Delta I)$ as defined in (d) above,

t_3 = 10 seconds

(f) Low reactor coolant loop flow - ≥ 90% of normal indicated loop flow as measured at elbow taps in each loop

(g) Low reactor coolant pump motor frequency - ≥ 57.5 Hz

(h) Reactor coolant pump under voltage - ≥ 70% of normal voltage

3. Other reactor trip settings

(a) High pressurizer water level - ≤ 92% of span

(b) Low-low steam generator water level - ≥5% of narrow range instrument span

(c) Low steam generator water level - ≥15% of narrow range instrument span in coincidence with steam/feedwater mismatch flow - ≤ 1.0×10^6 lbs/hr

(d) Turbine trip

(e) Safety injection - Trip settings for Safety Injection are detailed in TS Section 3.7.

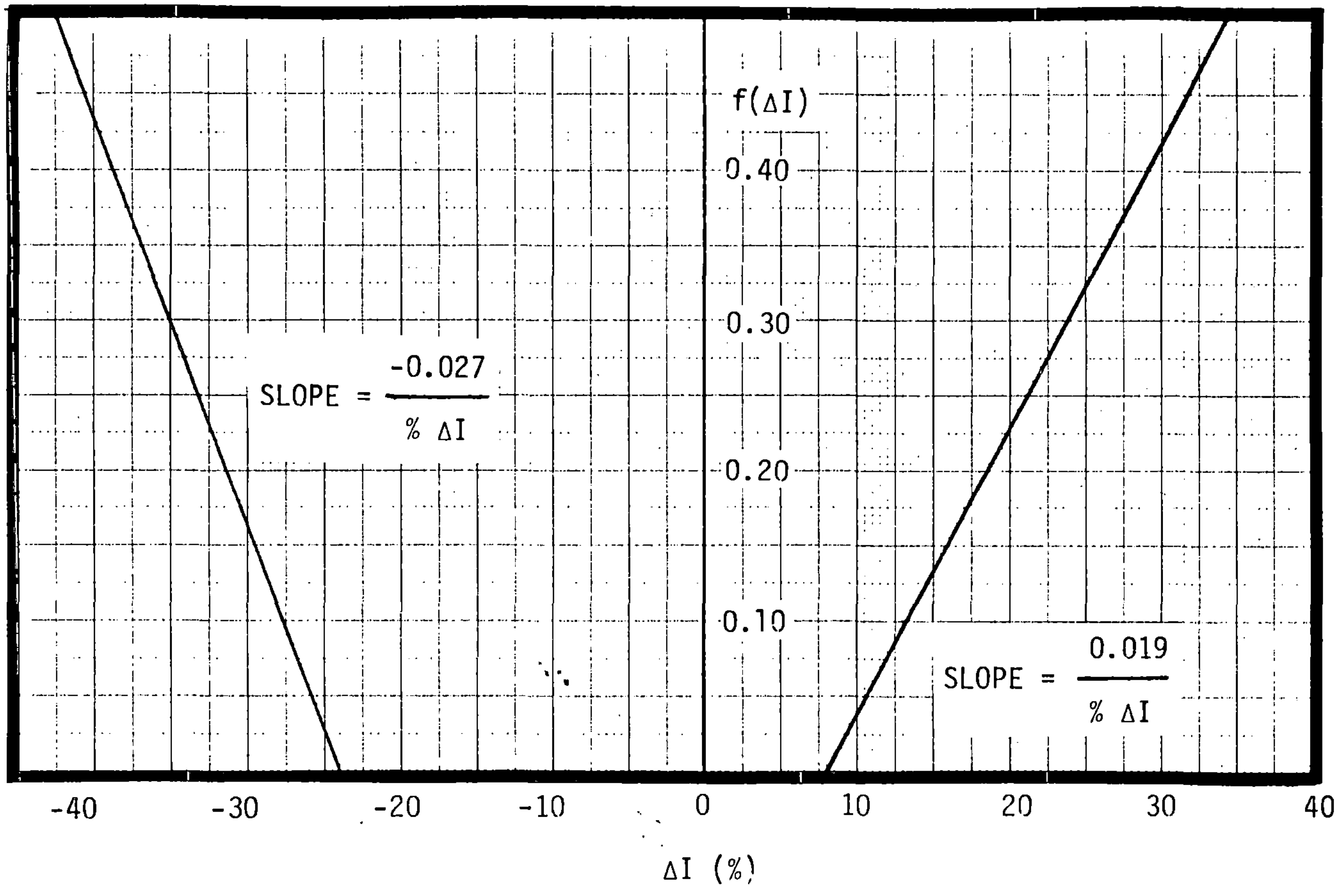


Figure 2.3-1 OPΔT and OTΔT f(ΔI) Function

B. Power Distribution Limits

1. At all times except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

   $F_Q(Z) \leq 2.18/P \times K(Z)$ for $P > 0.5$

   $F_Q(Z) \leq 4.36 \times K(Z)$ for $P \leq 0.5$

   $F^N_{\Delta H} \leq 1.55\ (1+0.3(1-P))$ for three loop operation

   $\leq 1.55\ (1+0.2(1-P))$ for two loop operation

   where P is the fraction of rated power at which the core is operating, K(Z) is the function given in TS Figure 3.12-8, and Z is the core height location of F_Q.

2. Prior to exceeding 75% power following each core loading and during each effective full power month of operation thereafter, power distribution maps using the movable detector system shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this confirmation:

   a. The measurement of total peaking factor F_Q^{Meas} shall be increased by eight percent to account for manufacturing tolerances, measurement error and the effects of rod bow. The measurement of enthalpy rise hot channel factor $F_{\Delta H}$ shall be increased by four percent to account for measurement error. If any measured hot channel factor exceeds its limit specified under Specification 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under Specification 3.12.B.1 are met. If the hot channel factors cannot be brought to within the limits of $F_Q(Z)$ 2.18 x K(Z) and $F^N_{\Delta H} \leq 1.55$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

It should be noted that the enthalpy rise factors are based on intergrals and are used as such in the DNB and LOCA calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in radial (x-y) power shapes throughout the core. Thus, the radial power shape at the point of maximum heat flux is not necessarily directly related to the enthalpy rise factors. The results of the loss of coolant accident analyses are conservative with respect to the ECCS acceptance criteria as specified in 10 CFR 50.46 using an upper bound envelope of 2.18 times the hot channel factor normalized operating envelope given by TS Figure 3.12-8.

When an F_Q measurement is taken, measurement error, manufacturing tolerances, and the effects of rod bow must be allowed for. Five percent is the appropriate allowance for measurement error for a full core map ($\geq$ 38 thimbles, including a minimum of 2 thimbles per core quandrant, monitored) taken with the movable incore detector flux mapping system, three percent is the appropriate allowance for manufacturing tolerances, and five percent is the appropriate allowance for rod bow. These uncertainties are statistically combined and result in a net increase of 1.08 that is applied to the measured value of F_Q.

In the specified limit of $F^N_{\Delta H}$ there is an eight percent allowance for uncertainties, which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55\ (1+0.3\ (1-P))/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods and can limit it to the desired value; he has no direct control over $F^N_{\Delta H}$, and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests and which may influence F_Q, can

ENCLOSURE 3

SAFETY EVALUATION FOR

RESTORATION OF CONTROL ROD

INSERTION LIMIT

CURVES FOR SURRY UNIT 1

SAFETY EVALUATION FOR

RESTORATION OF CONTROL ROD INSERTION LIMIT CURVES

FOR SURRY UNIT 1

The Control Rod Insertion Limit Curves as defined in the Technical Specifications limit control rod insertion during power operations to maintain the following parameters within previously analyzed limits: trip reactivity, shutdown margin, ejected rod worth and radial power peaking factors. During Unit 1 Cycle 7 operations, the insertion limits were required to be raised from the previously established limits (Figure 1) to the revised limits of Figure 2. This change was required to maintain the radial power peaking factors ($F^N_{\Delta H}$) below the Technical Specifications limits (i.e., $F^N_{\Delta H} \leq 1.55\ (1.0 + 0.2(1-P))$ where P = fraction of rated thermal power). The reload safety evaluation of Unit 1 Cycle 7[1] stated that all safety-related core characteristics are within the bounds of current accident analysis assumptions for either the previously established or the revised insertion limits, except for $F^N_{\Delta H}$. The reload evaluation further established that, after 1000 MWD/MTU of Cycle 7 burnup, the radial peaking factors fall within the limits defined by the relationship

$$F^N_{\Delta H} \leq 1.55\ (1.0 + 0.3\ (1-P))$$

when the previously established limits of Figure 1 are assumed. This calculation of $F^N_{\Delta H}$ versus power is depicted in Figure 3. Further, the Hot Zero Power peaking factors fall within these limits throughout Cycle 7 core life. Since it has been demonstrated[2] that these $F^N_{\Delta H}$ limits are acceptable for Surry when applied in conjunction with an appropriate set of core thermal limits and overtemperature/overpower ΔT setpoints, it is concluded that restoration of the Unit 1 rod insertion limits to the Figure 1 values after 1000 MWD/MTU of Cycle 7 operation does not result in an unreviewed safety question.

References:

1. Letter from W. L. Stewart (Vepco) to H. R. Denton (NRC) Serial No. 238, dated April 14, 1983, Attachment 2, "Reload Safety Evaluation for Surry Unit 1 Cycle 7 Redesigned Core".

2. "Safety Evaluation for a Revised $F_{\Delta H}^{N}$ Part Power Multiplier for Surry Units 1 and 2", Enclosure 1 to this submittal.

FIGURE 1

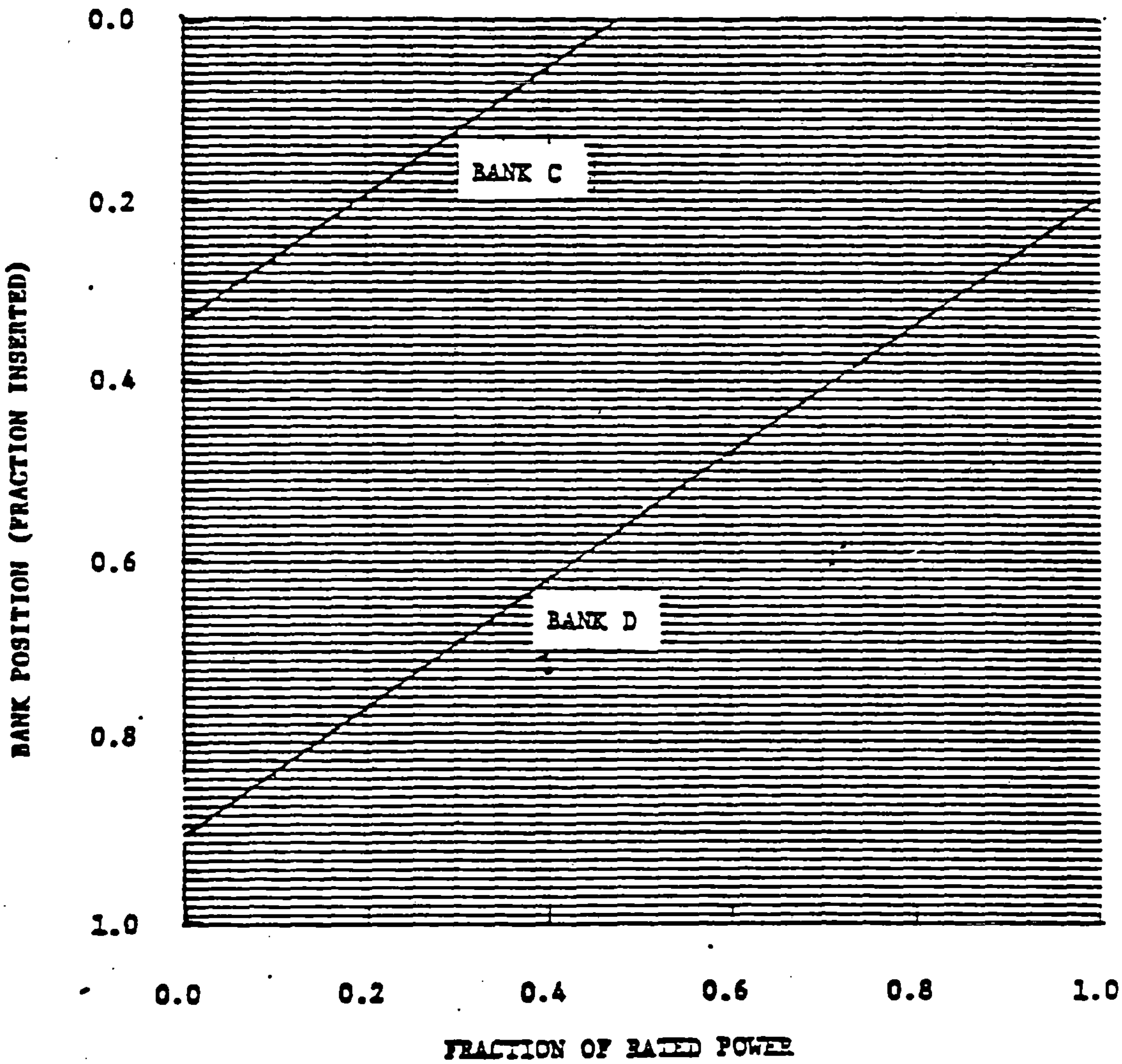


FIGURE 3.12-1A CONTROL BANK INSERTION LIMITS FOR 3-LOOP NORMAL OPERATION-UNIT 1

FIGURE 2

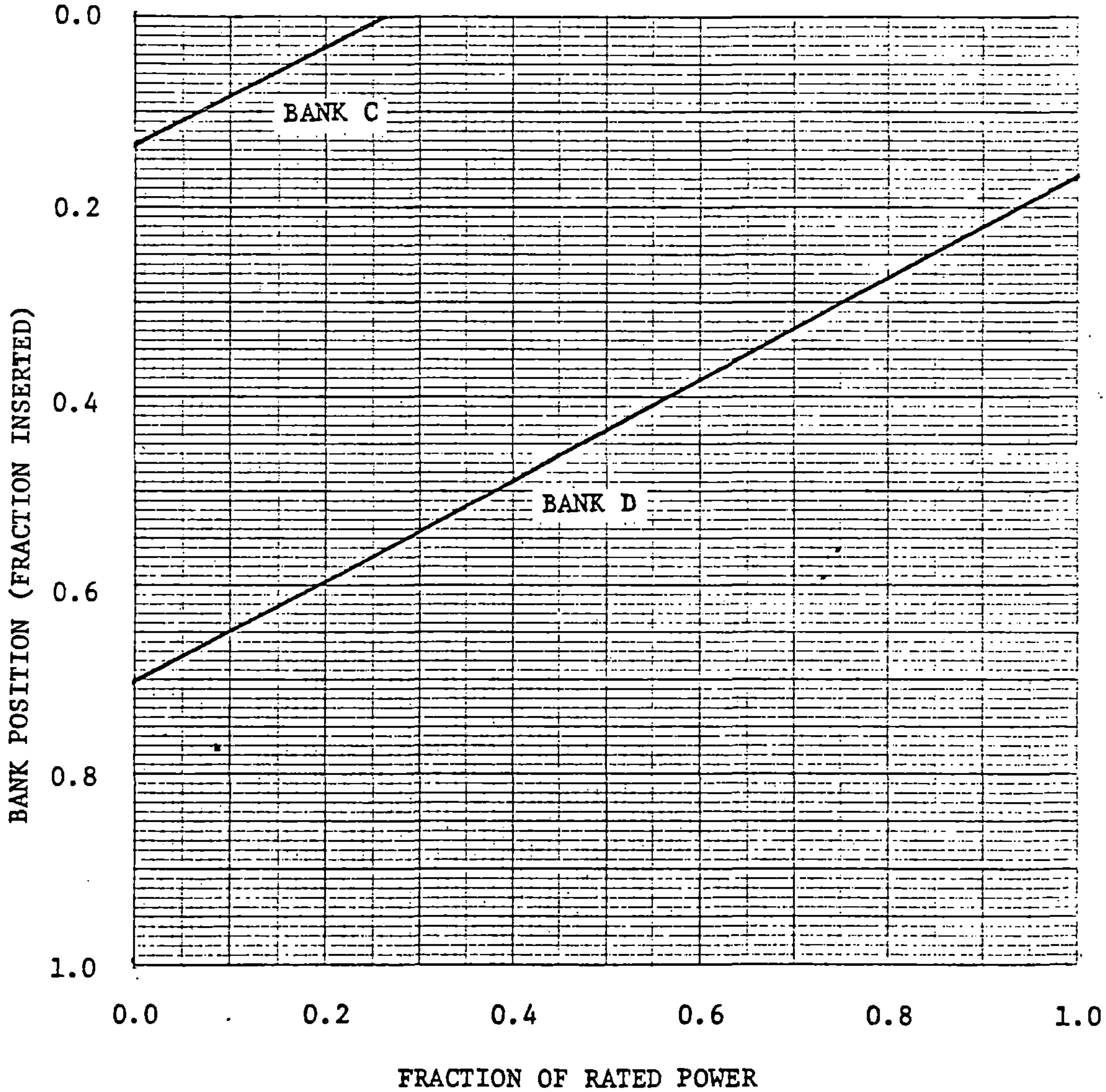
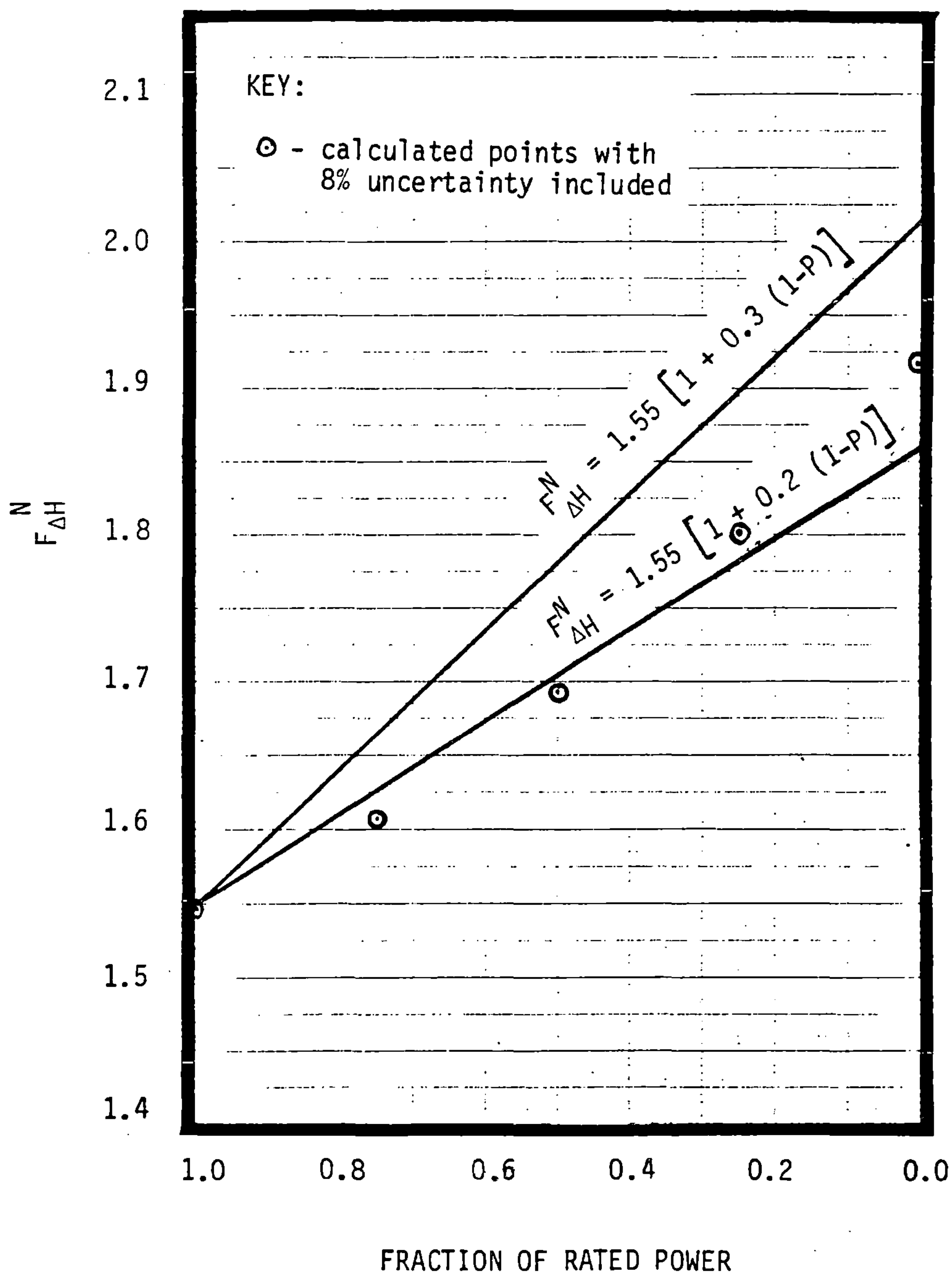


FIGURE 3.12-1A CONTROL BANK INSERTION LIMITS FOR 3-LOOP NORMAL OPERATION-UNIT 1

FIGURE 3

SURRY UNIT 1 CYCLE 7
CONSERVATIVE CALCULATION OF ENTHALPY RISE FACTOR*
WITH POWER LEVEL AT 1000 MWD/MTU
AND TECHNICAL SPECIFICATION LIMITS

*BASED ON FIGURE 1 INSERTION LIMITS



ENCLOSURE 4

SURRY 1 TECHNICAL SPECIFICATION CHANGES

on the maximum inserted rod worth in the unlikely event of a hypothetical assembly ejection and provide for acceptable nuclear peaking factors. The limit may be determined on the basis of unit startup and operating data to provide a more realistic limit which will allow for more flexibility in unit operation and still assure compliance with the shutdown requirement. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in the analysis of the hypothetical steam break accident. The rod insertion limits are based on end of core life conditions. The shutdown margin for the entire cycle length is established at 1.77% reactivity. All other accident analysis with the exception of the chemical and volume control system malfunction analysis are based on 1% reactivity shutdown margin. Relative positions of control rod banks are determined by a specified control rod bank overlap. This overlap is based on the consideration of axial power shape control. The specified control rod insertion limits have been established to limit the potential ejected rod worth in order to account for the effects of fuel densification. The various control rod assemblies (shutdown banks, control banks A, B, C, and D) are each to be moved as a bank; that is, with all assemblies in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuating pulses which shows the demand position of the banks, and a linear position indicator, Linear Variable Differential Transformer, which indicates the actual assembly position. The position indication accuracy of the Linear Differential Transformer is approximately ±5% of span (±12 steps) under steady state conditions. The relative accuracy of the linear position indicator has been considered in establishing the maximum allowable deviation of a control rod assembly from its indicated group step demand position. In the event that the linear position indicator is not

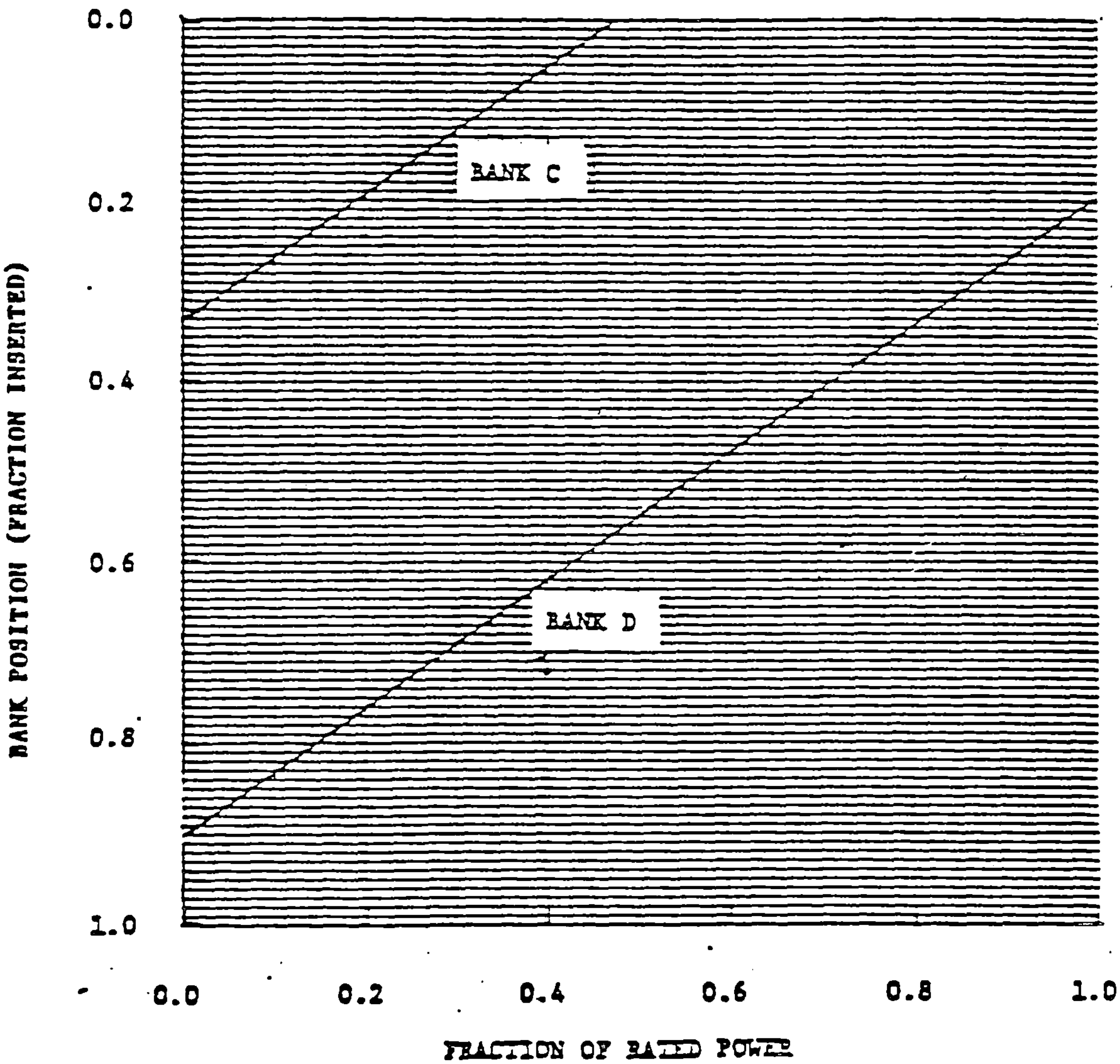


FIGURE 3.12-1A CONTROL BANK INSERTION LIMITS FOR 3-LOOP NORMAL OPERATION-UNIT 1

COMMONWEALTH OF VIRGINIA)
)
CITY OF RICHMOND)

The foregoing document was acknowledged before me, in and for the City and Commonwealth aforesaid, today by W. L. Stewart, who is Vice President-Nuclear Operations, of the Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 2nd day of May, 19 83.

My Commission expires: 2-26, 19 85.

Ann C. McRae
Notary Public

(SEAL)

M2/004