ATTACHMENT A

PROPOSED TECHNICAL SPECIFICATION AMENDMENTS

8308230353 830819
PDR ADOCK 05000250
P PDR

LIST OF FIGURES

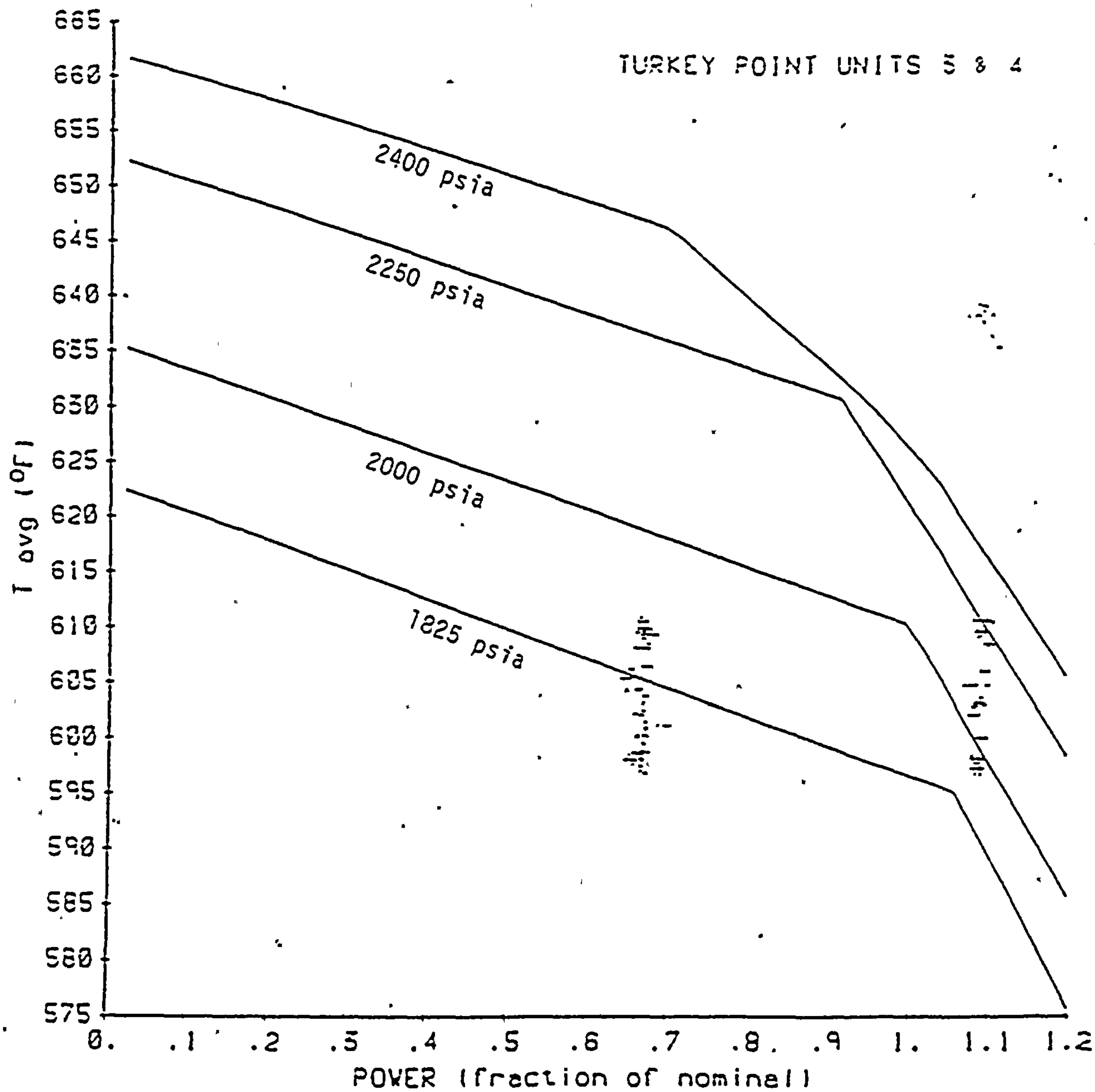


Figure 2.1-1 Reactor Core Thermal and Hydraulic
Safety Limits, Three Loop Operation

This Figure intentionally deleted.

Figure 2.1-1a

This Figure intentionally deleted.

Figure 2.1-1b

REACTOR COOLANT TEMPERATURE

Overtemperature $\Delta T \leq \Delta T_o \, [K_1 - 0.0107\,(T-574) + 0.000453\,(P-2235) - f\,(\Delta q)]$

ΔT_o = Indicated ΔT at rated power, F

T = Average temperature, F

P = Pressurizer pressure, psig

$f(\Delta q)$ = 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 startup tests such that:

For $(q_t - q_b)$ within + 10 percent and -14 percent 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, $f(\Delta q) = 0$.

For each percent that the magnitude of $(q_t - q_b)$ exceeds + 10 percent, the Delta-T trip setpoint shall be automatically reduced by 3.5 percent of its value at interim power.

For each percent that the magnitude of $(q_t - q_b)$ exceeds -14 percent, the Delta-T trip setpoint shall be automatically reduced by 2 percent of its value at interim power.

K_1 (Three Loop Operation) = 1.095

(Two Loop Operation) = 0.88

Overpower $\Delta T \leq \frac{T_o}{\Delta T_o} \left[1.09 - K_1 \frac{dT}{dt} - K_2 (T - T') - f(\Delta q)\right]$

ΔT_o = Indicated T at rated power, F

ΔT = Average temperature, F

T' = Indicated average temperature at nominal conditions and rated power, F

K_1 = 0 for decreasing average temperature; 0.2 sec./F for increasing average temperature

K_2 = 0.00068 for T equal to or more than T'; 0 for T less than T'

$\frac{dT}{dt}$ = Rate of change of temperature, F/sec

$f(\Delta q)$=As defined above.

Pressurizer

Low Pressurizer pressure - equal to or greater than 1835 psig.

High Pressurizer pressure - equal to or less than 2385 psig.

High Pressurizer water level - equal to or less than 92% of full scale.

Reactor Coolant Flow

Low reactor coolant flow - equal to or greater than 90% of normal indicated flow.

Low reactor coolant pump motor frequency equal to or greater than 56.1 Hz.

Undervoltage on reactor coolant pump motor bus - equal to or greater than 60% of normal voltage.

Steam Generators

Low-low steam generator water level - equal to or greater than 15% of narrow range instrument scale.

6. DNB PARAMETERS

The following DNB related parameter limits shall be maintained during power operation:

a. Reactor Coolant System Tavg $\leq$ 578.2°F

b. Pressurizer Pressure $\geq$ 2220 psia*

c. Reactor Coolant Flow $\geq$ 268,500 gpm

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce thermal power to less than 5% of rated thermal power using normal shutdown procedures.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours.

Compliance with c. is demonstrated by verifying that the parameter is within its limits after each refueling cycle.

*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.

reactivity insertion upon ejection greater than 0.3% $\Delta k/k$ at rated power. Inoperable rod worth shall be determined within 4 weeks.

b. A control rod shall be considered inoperable if

(a) the rod cannot be moved by CRDM, or
(b) the rod is misaligned from its bank by more than 15 inches, or
(c) the rod drop time is not met.

c. If a control rod cannot be moved by the drive mechanism, shutdown margin shall be increased by boron addition to compensate for the withdrawn worth of the inoperable rod.

5. CONTROL ROD POSITION INDICATION

If either the power range channel deviation alarm or the rod deviation monitor alarm is not operable, rod positions shall be logged once per shift and after a load change greater than 10% of rated power. If both alarms are inoperable for two hours or more, the nuclear overpower trip shall be reset to 93% of rated power.

6. POWER DISTRIBUTION LIMITS

a. Hot channel factors:

(1) F_Q Limit

The hot channel factors (defined in Bases) must meet the following limits at all times except during low power physics tests:

$$F_Q(Z) \leq ([F_Q]_L/P) \times K(Z), \text{ for } P \geq 0.5$$

$$F_Q(Z) \leq (2 \times [F_Q]_L) \times K(Z), \text{ for } P \leq 0.5$$

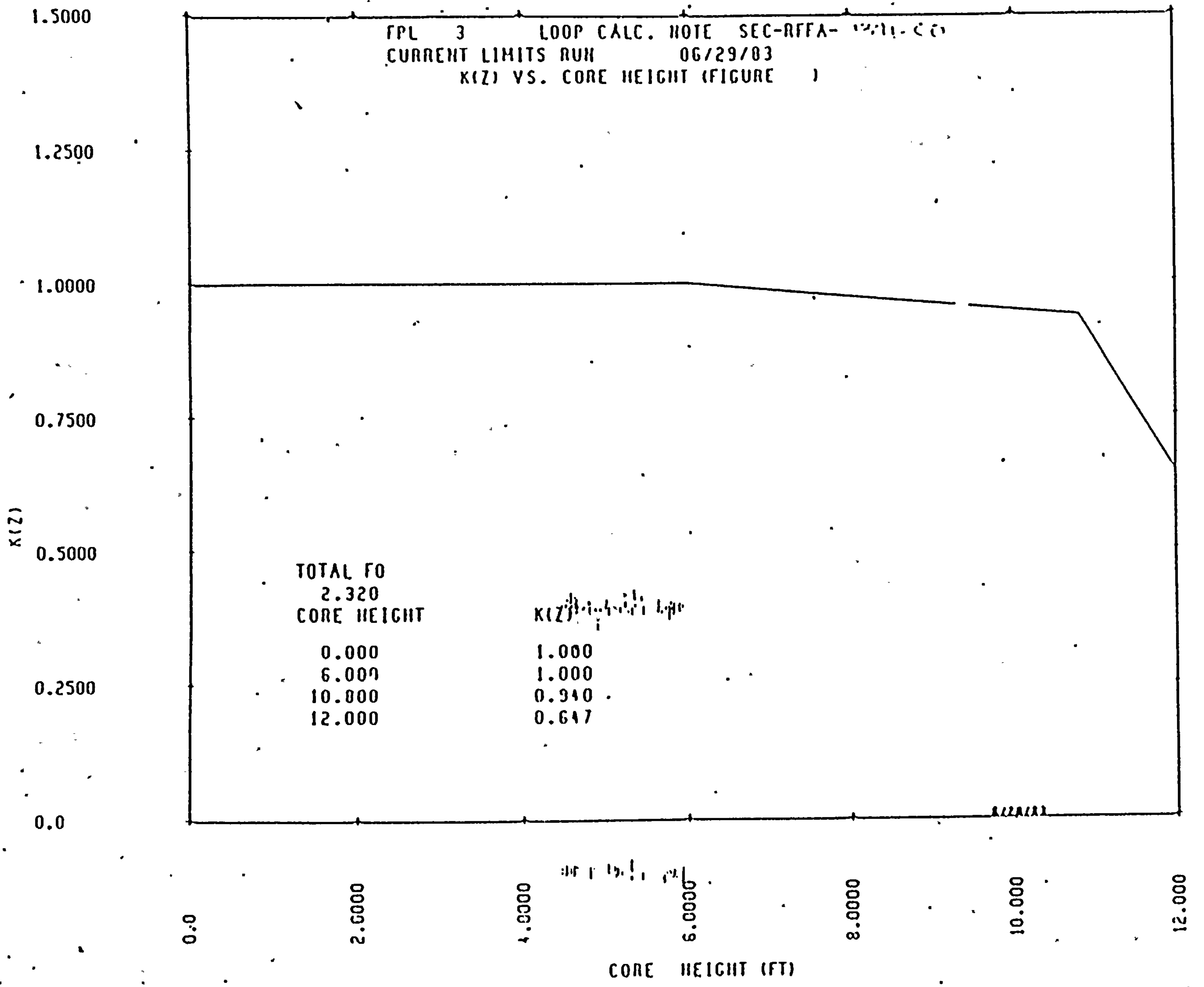
$$F^N_{\Delta H} \leq 1.62\ [1.0 + 0.3\ (1 - P)]$$

Where P is the fraction of rated power at which the core is operating; K(Z) is the function given in Figure 3.2-3; Z is the core height location of F_Q.

Plugging level	$[F_Q]_L$	Figure Number for K(Z)
$\leq$ 5%	2.32	3.2-3

(2) Augmented Surveillance (MIDS)

If $[F_Q]_P$, as predicted by approved physics calculations, exceeds $[F_Q]_L$ then the power will be limited to a turnon power fraction, P_T, equal to the ratio of $[F_Q]_L$ divided by $[F_Q]_P$, or, for operation at power levels above P_T, augmented surveillance of hot channel factors shall be implemented, except in Base Load



FPL 3 LOOP CALC. NOTE SEC-RFFA-
CURRENT LIMITS RUN 06/29/83
K(Z) VS. CORE HEIGHT (FIGURE)
1.5000
1.2500
1.0000
0.7500
0.5000
0.2500
0.0
K(Z)
TOTAL FQ
2.320
CORE HEIGHT
K(Z)
0.000
6.000
10.000
12.000
1.000
1.000
0.940
0.647
0.0
2.0000
4.0000
6.0000
8.0000
10.000
12.000
CORE HEIGHT (FT)

FIGURE 3.2-3

This Figure intentionally deleted.

Figure 3.2-3a

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

The curves are conservative for an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.62 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:

$$F^N_{\Delta H} \leq 1.62 \, [1 + 0.3 \, (1-P)]$$

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 limit assuming the axial power imbalance is within the limits of the $f(\Delta q)$ 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.

Fuel rod bowing reduces the values of DNB ratio (DNBR). The amount of the DNBR reduction is 4.7% for LOPAR fuel with the L-grid DNB correlation and 5.5% for the OFA fuel with the WRB-1 DNB correlation. The penalties are calculated pursuant to "Fuel Rod Bow Evaluation," WCAP-8691-P-A, Rev. 1 (proprietary) and WCAP-8692 Rev. 1 (non-proprietary). The restrictions of the Core Thermal Hydraulic Safety Limits assure that an amount of DNBR margin greater than or equal to the above penalties is retained to offset the rod bow DNBR penalty.

An upper bound envelope as defined by normalized peaking factor axial dependence of Figure 3.2-3, has been determined to be consistent with the technical specifications on power distribution control as given in Section 3.2.

The results of the loss of coolant accident analyses based on this upper bound envelope indicate a peak clad temperature could theoretically exceed the 2200°F limits. To ensure the criteria are not violated, MIDS will be used to provide a more exact indication of F_Q. Note that MIDS and a penalty on F_Q are only required above P_T to meet the acceptance criteria as justified in the analyses. Below P_T, the nuclear analyses of credible power shapes consistent with these specifications have shown that the limit of $[F_Q]_L/P$ times Figure 3.2-3 is not exceeded provided the limits of Figure 3.2-3 are applied.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. These uncertainties only apply if the map is taken for purposes other than the determination of P_{BL} and P_{RB}.

In the specified limit of $F^N_{\Delta H}$, there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \le 1.62/1.08$. 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) although 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 prediction for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of start-up physics tests, at least once each full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear